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Technical Specifications

ORIGINAL

Beaver Valley Power Station, Unit 2

Docket No. 50-412

Appendix "A" to
License No. NPF-73

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

August 1987

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1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2652 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principal specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related safety function(s).

REPORTABLE EVENT

1.7 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

CONTAINMENT INTEGRITY (Continued)

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed.

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3., and

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2.

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one (1) excore detector inoperable, the remaining three (3) detectors shall be used for computing the average.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Regulatory Guide 1.109, 1977 or TID 14844.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

STAGGERED TEST BASIS (continued)

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

b. The testing of one (1) system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.23 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

AXIAL FLUX DIFFERENCE

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two-section excore neutron detector.

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

$\bar{E}$ - AVERAGE DISTINTEGRATION ENERGY

1.26 $\bar{E}$ shall be the average sum (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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SOURCE CHECK

1.27 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

PROCESS CONTROL PROGRAM

1.28 A PROCESS CONTROL PROGRAM (PCP) shall be the manual or set of operating parameters detailing the program of sampling, analysis, and evaluation by which SOLIDIFICATION of wet radioactive wastes is assured. Requirements of the PCP are provided in Specification 6.14.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet radioactive wastes into a form that meets shipping and burial ground requirements.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints. Requirements of the ODCM are provided in Specification 6.15.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is required to purify the confinement.

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VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

MAJOR CHANGES

1.35 MAJOR CHANGES to radioactive waste systems, as addressed in Paragraph 6.16.2, (liquid, gaseous and solid) shall include the following:

1) Major changes in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2) Major changes in the design of radwaste treatment systems (liquid, gaseous, and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions).

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries and persons who traverse portions of the site as the consequence of a public highway, railway, or waterway located within the confines of the site boundary. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥0.99	>5%	≥350°F
2. STARTUP	≥0.99	≤5%	≥350°F
3. HOT STANDBY	<0.99	0	≥350°F
4. HOT SHUTDOWN	<0.99	0	350°F >T_{avg} >200°F
5. COLD SHUTDOWN	<0.99	0	≤200°F
6. REFUELING**	≤0.95	0	≤140°F

---

*Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not applicable.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 3-loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4, and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

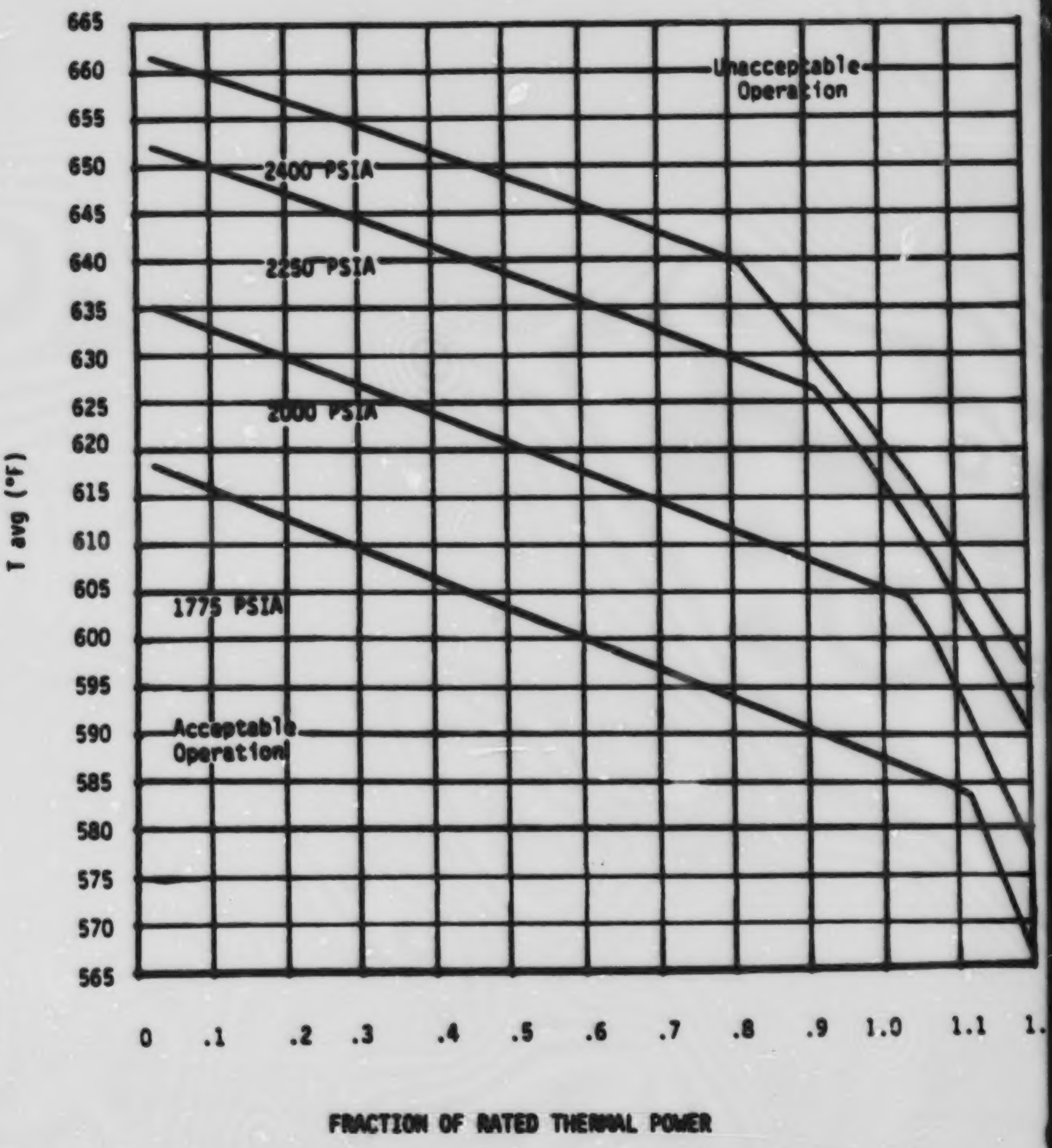

FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1 adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

EQUATION 2.2-1 $Z + R + S \leq TA$

where:

Z = The value for column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value of Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance in % of span) of Table 2.2-1 for the affected channel.

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TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	$\leq$ 109% of RTP*	$\leq$ 111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	$\leq$25% of RTP*	$\leq$27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.50	0	$\leq$ 5% of RTP* with a time constant $\geq$ 2 seconds	$\leq$ 6.3% of RTP* with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.50	0	$\leq$ 5% of RTP* with a time constant $\geq$ 2 seconds	$\leq$ 6.3% of RTP* with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	$\leq$ 25% of RTP*	$\leq$ 30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	7.3	5.18	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.8	1.38	1.72	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.67	$\geq$ 1945 psig***	$\geq$ 1935 psig***
10. Pressurizer Pressure-High	6.2	4.96	0.67	$\leq$ 2375 psig	$\leq$ 2383 psig
11. Pressurizer Water Level-High	8.0	2.18	1.67	$\leq$ 92% of instrument span	$\leq$ 93.8% of instrument span
12. Loss of Flow	2.5	2.01	0.60	$\geq$ 90% of loop design flow**	$\geq$ 89.5% of loop design flow**

---

* = RATED THERMAL POWER

** Loop design flow = 88,500 gpm

*** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to those values.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level-Low-Low	15.5	14.1[illegible]	1.67	≥ 15.5% of narrow range instrument span-each steam generator	≥ 14.7% of narrow range instrument span-each steam generator
14. a. Steam/Feedwater Flow Mismatch	20.0	2.66	See Note 6	≤ 40% of full steam flow at RTP*	≤ 43.1% of full steam flow at RTP*
b. Coincident with Steam Generator Water Level Low	25.0	2.18	1.67	≥ 25% of narrow range instrument span	≥ 23.2% of narrow range instrument span
15. Undervoltage - Reactor Coolant Pumps	27.7	1.39	0	≥75% of BUS Voltage-each bus	≥73% of BUS Voltage - each bus
16. Underfrequency-Reactor Coolant Pumps	10.0	0.20	0	≥57.5 Hz-each bus	≥57.4 Hz-each bus
17. Turbine Trip					
a. Emergency Trip Header Low Pressure	N.A.	N.A.	N.A.	≥1000 psig	≥958 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A	N.A.
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	N.A.	N.A.
20. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.
22. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	$\leq 30\%$ of RTP*	$\leq 32.1\%$ of RTP*
c. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	$\leq 49\%$ of RTP*	$\leq 51\%$ of RTP*
d. Power Range Neutron Flux, P-10 (Input to P-7)	N.A.	N.A.	N.A.	10% of RTP*	$\geq 7.9\%$ and $\leq 12.1\%$ of RTP*
e. Turbine Impulse Chamber Pressure, P-13 (Input to P-7)	N.A.	N.A.	N.A.	$<10\%$ of RTP* Turbine impulse pressure equivalent	$<12.1\%$ of RTP* Turbine impulse pressure equivalent

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau^1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T\left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1 (\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation;

$1 + \tau S$ $\frac{1}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1 , τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.28;

K_2 = 0.017/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in lead-lag compensator for T_{avg}, $\tau_4 = 30$ s, $\tau_5 = 4$ s;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

T = Average temperature, °F;

$\frac{1}{1+\tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

T' = ≤ 576.2°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.00082;

P = Pressurizer Pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -33% and +9%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -33%, the ΔT Trip Setpoint shall be automatically reduced by 2.52% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude $q_t - q_b$ exceeds +9%, the ΔT Trip Setpoint shall be automatically reduced by 1.75% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.7% of ΔT span.

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau^1 S)}{(1 + \tau_2 S)} \left(\frac{1}{(1 + \tau_3 S)}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{(1 + \tau_7 S)}\right) \left(\frac{1}{(1 + \tau_6 S)}\right) T - K_6 [T \left(\frac{1}{(1 + \tau_6 S)}\right) - T''] - f_2(\Delta i)\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_4 = 1.0781;

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature;

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constant utilized in rate-lag compensator for T_{avg}, $\tau_7 = 10$ s;

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

K_6 = 0.0012/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;

T = Average Temperature, °F;

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$576.2°F);

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.6% of ΔT span.

NOTE 5: The sensor error for temperature is 1.72% and 0.73% of span for pressure.

NOTE 6: The sensor error for steam flow is 1.0%, for feedwater flow is 1.0%, and for steam pressure is 0.83% of span.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 R-Grid correlation. The W-3 R-Grid DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve of Figure 2.1-1 shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55 [1 + 0.3 (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f(\Delta I)$ function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trip will reduce the setpoint to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this safety limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the reactor trip setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S < TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above 1.30 for control rod drop accidents. At high power a multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than 1.30.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The intermediate range channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. Although no explicit credit was taken for operation of the Source Range Channels in the accident analyses, operability requirements in the Technical Specifications will ensure that the Source Range Channels are available to mitigate the consequences of an inadvertent control bank withdrawal in MODES 3, 4 and 5.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam line breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Release."

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Pressure trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and an increasing power, automatically reinstated by P.7.

On decreasing power, the low setpoint trip is automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, automatically reinstated by P-7.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. On decreasing power, the pressurizer high water level trip is automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, automatically reinstated by P-7. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 10 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90 percent of nominal full loop flow. Above 30 percent (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90 percent of nominal full loop flow.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by $\geq 1.55 \times 10^6$ lbs/hour in any loop. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument.

These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.6 seconds.

On decreasing power, the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 Above the setpoint P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. Below the setpoint source range level trips are automatically reactivated and high voltage restored.

P-7 Above the setpoint P-7 automatically enables reactor trips on low flow or coolant pump breaker open in more than one primary coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. Below the setpoint the above listed trips are automatically blocked.

P-8 Above the setpoint P-8 automatically enables reactor trip on low flow in one or more primary coolant loops. Below the setpoint P-8 automatically blocks the above listed trip.

P-9 Above the setpoint P-9 automatically enables a reactor trip on turbine trip. Below the setpoint P-9 automatically blocks a reactor trip on turbine trip.

P-10 Above the setpoint P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. Below the setpoint the Intermediate Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements requirements. This provision shall not prevent passage through OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 2 hours, action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply, by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in MODES 5 or 6.

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

APPLICABILITY

SURVEILLANCE REQUIREMENTS

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days.

c. The provision of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥1.77% Δk/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN <1.77% Δk/k, immediately initiate and continue boration at >30 gpm of > 7000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥1.77% Δk/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2,# at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2,## at least once during control rod withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

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*See Special Test Exception 3.10.1

#With $K_{eff} \geq 1.0$

##With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

1. Reactor Coolant System boron concentration,
2. Control rod position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

f. The Reactor Coolant System shall be borated to at least the COLD SHUTDOWN boron concentration prior to manually blocking the Low Pressurizer Pressure Safety Injection Signal and shall remain at this boron concentration or greater at all times during which this signal is blocked.

4 1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 1.0% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN < 1.0% Δk/k, immediately initiate and continue boration at $\geq$ 30 gpm of $\geq$ 7000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1.0% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

    1. Reactor Coolant System boron concentration,
    2. Control rod position,
    3. Reactor Coolant System average temperature,
    4. Fuel burnup based on gross thermal energy generation,
    5. Xenon concentration, and
    6. Samarium concentration.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the core shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: All MODES.

ACTION:

With the flow rate of reactor coolant through the core < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the core shall be determined to be $\geq$ 3000 gpm prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one RHR pump is in operation and supplying $\geq$ 3000 gpm through the core.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT (MTC)

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Moderator Temperature Coefficient (MTC) shall be:

a. Less positive than 0×10^{-4} $\Delta k/k/{}^{\circ}F$,

b. Less negative than -5.0×10^{-4} $\Delta k/k/{}^{\circ}F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the Moderator Temperature Coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

---

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.3.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 541°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2.*#

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be > 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F with the (T_{avg}) deviation alarm not reset.

---

*See Special Test Exception 3.10.3.

#With $K_{eff} \geq 1.0$.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid storage system via a boric acid transfer pump to a charging pump to the Reactor Coolant System if only the boric acid storage tank is OPERABLE as given in Specification 3.1.2.7.a for MODES 5 and 6 or as given in Specification 3.1.2.8.a for MODE 4; or

b. The flow path from the refueling water storage tank via a charging pump or a low head safety injection pump (with an open RCS vent of greater than or equal to 3.14 square inches) to the Reactor Coolant System if the refueling water storage tank is OPERABLE as given in Specification 3.1.2.7.b for MODES 5 and 6 or as given in Specification 3.1.2.8.b for MODE 4.

APPLICABILITY: MODES 4, 5 and 6

ACTION

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

   2. Verifying that the temperature of the heat traced portion of the flow path is $\geq$ 65°F when a flow path from the boric acid tanks is used and the ambient air temperature of the Auxiliary Building is < 65°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and one charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via one charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

2. Verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is $\geq$ 65°F when the ambient air temperature of the Auxiliary Building is $<$ 65°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown by cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 or Low Head Safety Injection Pump (with an open Reactor Coolant System vent of greater than or equal to 3.14 square inches) shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 4, 5 and 6

ACTION:

With none of the above pumps OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump or Low Head Safety Injection pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a differential pressure of $\geq$ 2437 psid when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, except the above required charging pump, shall be demonstrated inoperable* by verifying that the control switches are placed in the PULL-TO-LOCK position and tagged within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first, and at least once per 12 hours thereafter.

4.1.2.3.3 When the Low Head Safety Injection pump is used in lieu of a charging pump, the Low Head Safety Injection pump shall be demonstrated OPERABLE by:

a. Verification of an OPERABLE RWST pursuant to 4.1.2.7 and 4.1.2.8
b. Verification of an OPERABLE Low Head Safety Injection Pump pursuant to Specification 4.5.2.b.2,
c. Verification of an OPERABLE Low Head Safety Injection flow path from the RWST to the Reactor Coolant System once per shift, and
d. Verification that the vent is open at least once per 12 hours.**

---

*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

**Except when the vent path is provided with a valve which is locked or provided with remote position indication, or sealed, or otherwise secured in the open position, then verify these valves open at least once per 7 days.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a differential pressure of $\geq$ 2437 psid when tested pursuant to Specification 4.0.5.

---

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 One boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path thru the boric acid transfer pump of Specification 3.1.2.1.a, is OPERABLE.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1.a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid transfer pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid transfer pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a differential pressure of $\geq$ 102 psid when tested pursuant to Specification 4.0.5.

BORIC ACID TRANSFER PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2.a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2.a is OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With no boric acid transfer pump OPERABLE, restore at least one boric acid transfer pump to OPERABLE STATUS within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F; restore at least one boric acid transfer pump to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a differential pressure of $\geq$ 102 psid when tested pursuant to Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

   1. A minimum contained volume of 2315 gallons,
   2. Between 7000 and 7700 ppm of boron, and
   3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

   1. A minimum contained volume of 217,000 gallons,
   2. A minimum boron concentration of 2000 ppm, and
   3. A minimum solution temperature of 45°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration of the water,
   2. Verifying the water level of the tank, and
   3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is < 45°F

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2.

a. A boric acid storage system with:

   1. A minimum contained volume of 13,390 gallons,
   2. Between 7000 and 7700 ppm of boron, and
   3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

   1. A minimum contained volume of 859,248 gallons,
   2. A boron concentration between 2000 and 2100 ppm, and
   3. A solution temperature of $\geq$ 45°F and $\leq$ 50°F.

APPLICABILITY: MODES 1, 2, 3 & 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

   1. Verifying the boron concentration in each water source,

   2. Verifying the water level in each water source, and

   3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the RWST ambient air temperature is > 50°F or < 45°F.

ISOLATION OF UNBORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.9 Provisions to limit flow capability from unborated water sources to the reactor coolant system to less than or equal to 85 gpm by means of a flow limiting orifice shall be OPERABLE.

APPLICABILITY MODES 4 and 5.

ACTION

With the requirements of the above specification not satisfied immediately suspend all operations involving positive reactivity changes and, if within 1 hour the required SHUTDOWN MARGIN is not verified, initiate and continue boration as specified in Specification 3.1.1.1 for MODE 4 and 3.1.1.2 for MODE 5 until the SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.2.9 The provisions to limit flow capability to less than or equal to 85 gpm from unborated water sources to the reactor coolant system shall be determined to be OPERABLE at least once per 31 days by verifying that valve 2CHS-37 is closed.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position, as determined in accordance with Specification 3.1.3.2) corresponding to their respective group demand counter position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group demand counter position by more than ± 12 steps (indicated position determined in accordance with Specification 3.1.3.2), be in HOT STANDBY within 6 hours.

c. With one full length rod trippable but inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group demand counter position by more than ± 12 steps (indicated position determined in accordance with Specification 3.1.3.2), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the hour and, within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

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*See Special Test Exceptions 3.10.2 and 3.10.3

LIMITING CONDITION FOR OPERATION (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 Each shutdown and control rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

4.1.3.1.2 Each full length rod position shall be determined to be within ± 12 steps of the associated group demand counter by verifying the individual rod position at least once per 12 hours except during intervals when the Rod Position Deviation monitor is inoperable, then verify the group position at least once per 4 hours.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracked Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Systems Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*#, 4*# and 5*#

ACTION:

With less than the above required group demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at least once per 18 months.

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*With the reactor trip system breakers in the closed position.

#See Special Test Exceptions Specification 3.10.5.

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REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$ 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 541°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head.

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification (4.1.3.1.1), within one hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification (3.1.3.1).

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. At least once per 24 hours thereafter.

---

*See Special Test Exception 3.10.2 and 3.10.3

#With Keff $\geq$ 1.0

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.1, either:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figure, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

---

*See Special Test Exception 3.10.2 and 3.10.3

#with Keff $\geq$ 1.0

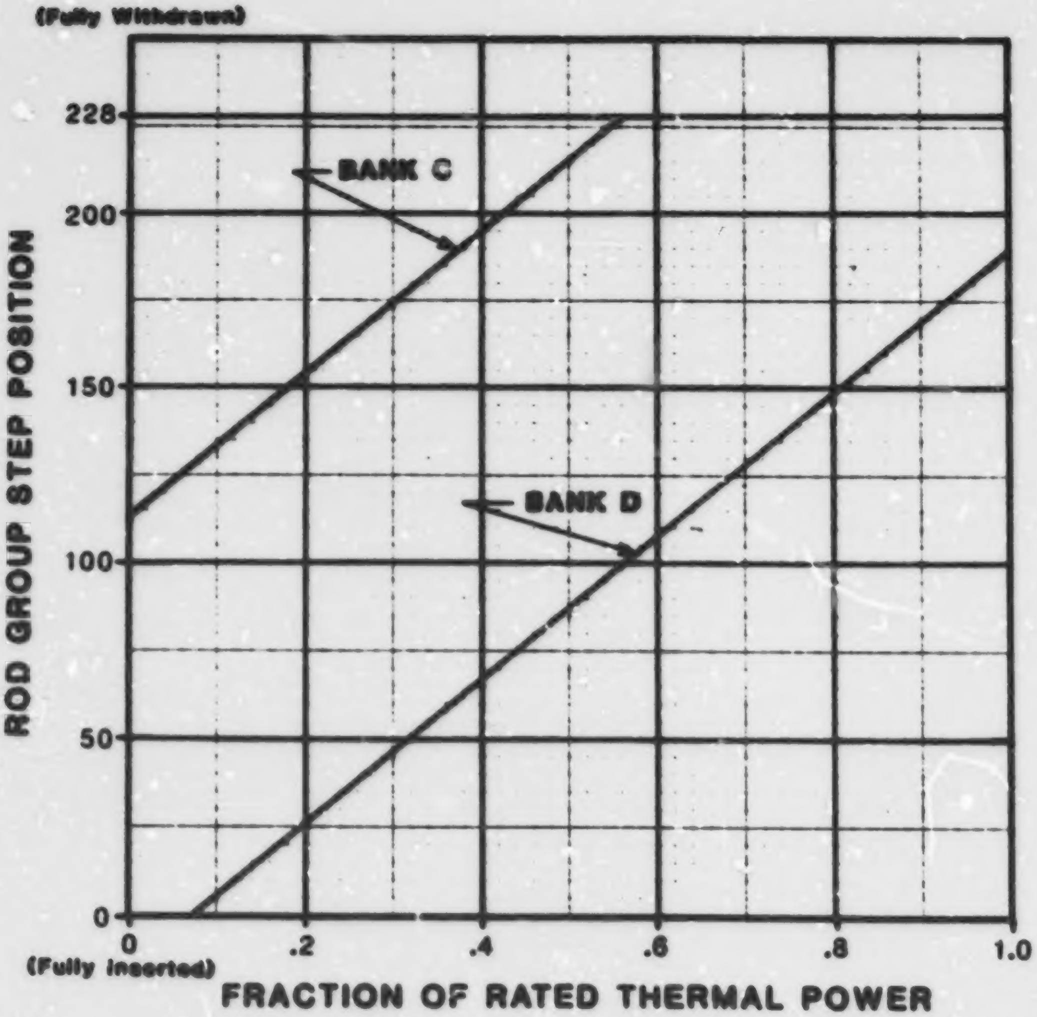

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FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER THREE LOOP OPERATION

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3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±7 percent target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 above 50 Percent RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ± 7 percent target band about the target flux difference and with THERMAL POWER:

1. Above 90 percent of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90 percent of RATED THERMAL POWER.

2. Between 50 percent and 90 percent of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the ± 7 percent target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55 percent of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

---

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. THERMAL POWER shall not be increased above 90 percent of RATED THERMAL POWER unless the indicated AFD is within the ± 7 percent target band and ACTION a.2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50 percent of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 7 percent target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15 percent of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist durin the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 7 percent target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. POWER OPERATION outside of the ± 7 perce target band shall be accumulated on a time basis of:

a. One-minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50 percent of RATED THERMAL POWER, and

b. One-half-minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

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POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

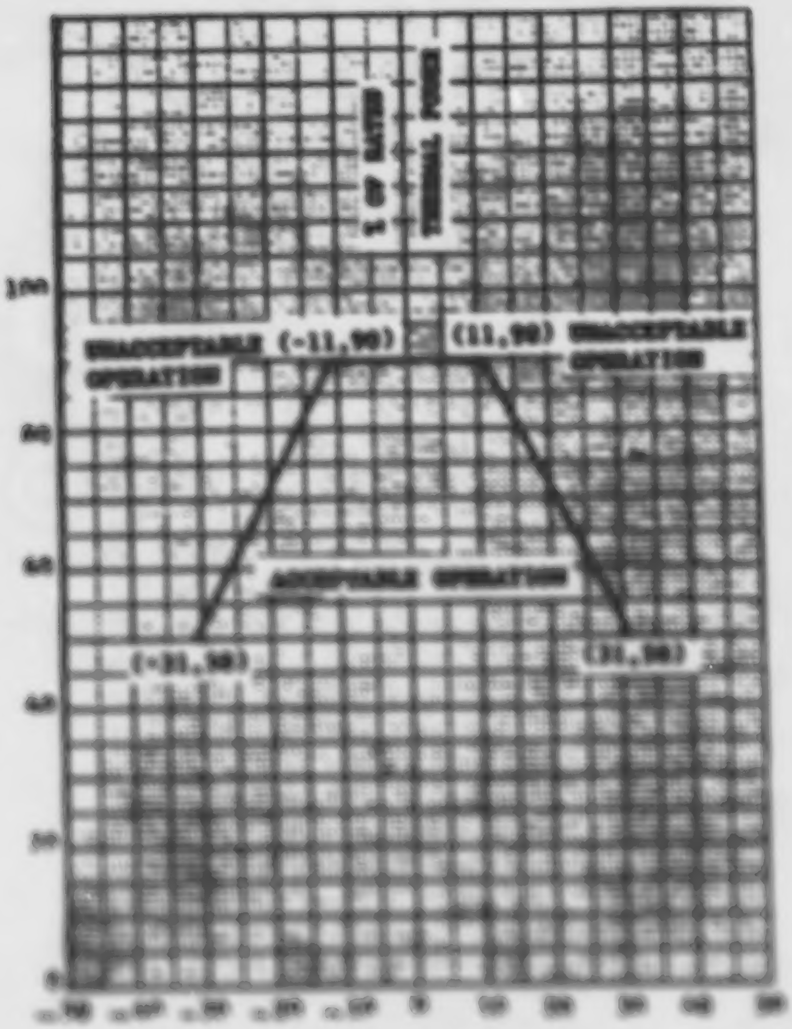

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1 ercent for each 1 percent $F_Q(Z)$ exceeds the limit within 15 minues and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for p to a total of 72 hours; subsequent POWER OPERATION may procee provided the Overpower ΔT Trip Setpoints have been reduced at last 1 percent for each 1 percent $F_Q(Z)$ exceeds the limit. The Ovrpower ΔT Trip Setpoint reduction shall be performed with the reactor subcritical.

b. Identify and correct the cause o the out of limit condition prior to increasing THERMAL POWER; THERML POWER may then be increased provided $F_Q(Z)$ is demonstrated though incore mapping to be within its limit.

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SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5 percent of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5 percent to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in b, above to:

   1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

   2. The relationship:

   $$F_{xy}^{L} = F_{xy}^{RTP} \quad [1+0.2(1-P)]$$

   where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

   1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L}:

      a) Either within 24 hours after exceeding by 20 percent of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

      b) At least once per 31 EFPD, whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.14.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15 percent, inclusive.
2. Upper core region from 85 to 100 percent inclusive.
3. Grid plane regions of core height (± 2.88 inches) measured from grid centerline.
4. Core plane regions within ± 2 percent of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3 percent to account for manufacturing tolerances and further increased by 5 percent to account for measurement uncertainty.

K(Z) - NORMALIZED $F_Q(Z)$
AS A FUNCTION OF CORE HEIGHT

3-LOOP
BEAVER VALLEY - UNIT 2

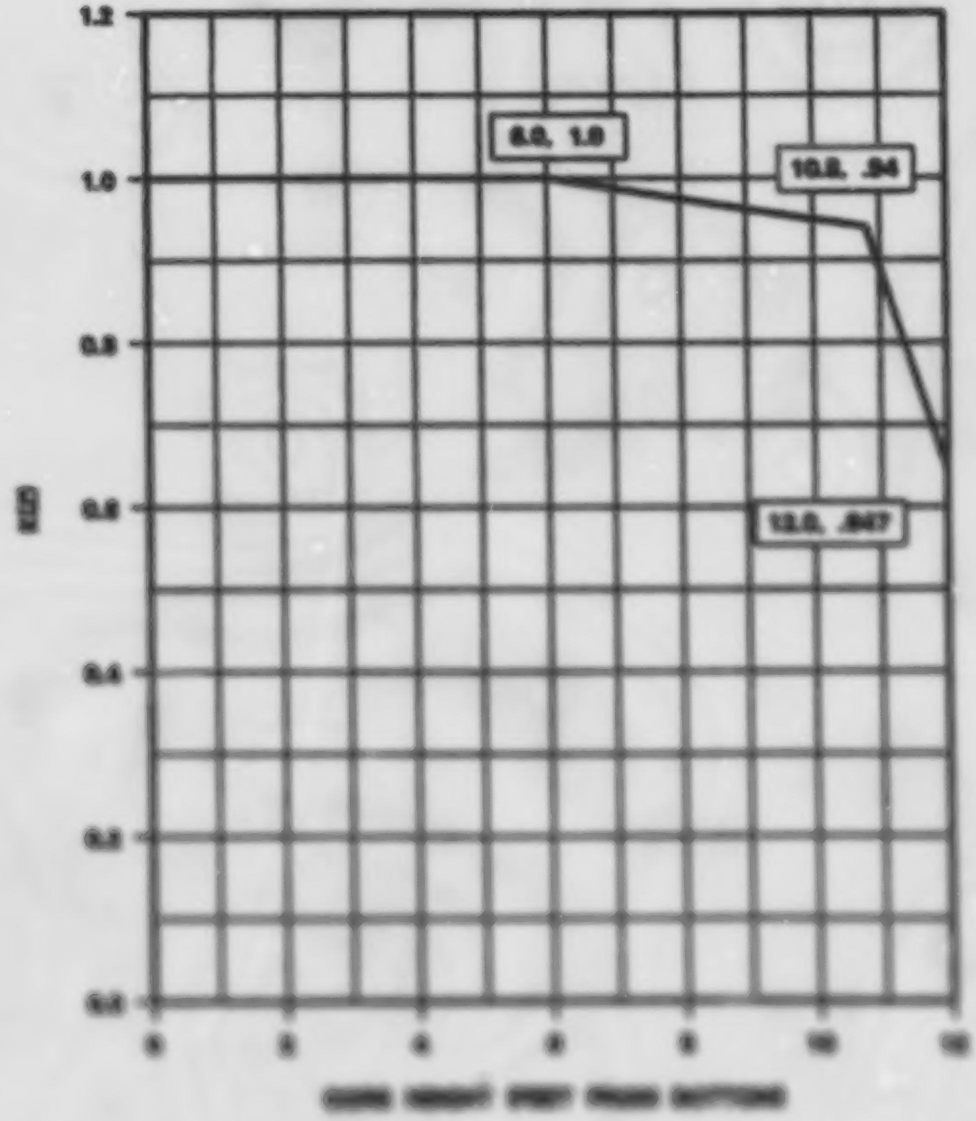

FIGURE [illegible]

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POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER, subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50 percent of RATED THERMAL POWER prior to exceeding this THERMAL power, at a nominal 75 percent of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95 percent or greater RATED THERMAL POWER.

[illegible]

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75 percent of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

POWER DISTRIBUTION LIMITS

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 above 50 Percent OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but ≤ 1.09:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3 percent for each 1 percent of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50 percent of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Reduce THERMAL POWER at least 3 percent for each 1 percent of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55 percent of RATED THERMAL POWER within the next 4 hours.

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*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50 percent of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50 percent of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER > 75 percent of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1*

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1.1 Each of the parameters of Table 3.2-1 shall be verified to be indicating within their limits at least once per 12 hours.

4.2.5.1.2 The provisions of Specification 4.0.3 and 4.0.4 are not applicable for the reactor startups following the initial fueling for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total flow rate indicators.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

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*The provisions of Specification 3.0.4 are not applicable for the reactor startup following the initial fueling for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total [illegible]

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 580.2°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq$ 274,800 gpm**

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*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel and interlock and automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements* during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by interlock operation. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

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*For the automatic trip logic, the surveillance requirements shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output including, as a minimum, a continuity check of output devices.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Reactor Trip	2	1	2	1, 2, 3*, 4*, and 5*	12
2.	Power Range, Neutron Flux					
	a. High Setpoint	4	2	3	1, 2	2
	b. Low Setpoint	4	2	3	$1^{(1)}$, 2	2
3.	Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4.	Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
5.	Intermediate Range, Neutron Flux	2	1	2	$1^{(1)}$, 2, 3*, 4*, and 5*	3
6.	Source Range, Neutron Flux (Below P-10)					
	a. Startup	2	1	2	$2^{(2)}$, 3*, 4*, and 5*	4
	b. Shutdown	2	1	2	3, 4 and 5	5
7.	Overtemperature ΔT Three Loop Operation	3	2	2	1, 2	7
8.	Overpower ΔT Three Loop Operation	3	2	2	1, 2	7
9.	Pressurizer Pressure-Low (Above P-7)	3	2	2	1, 2	7
10.	Pressurizer Pressure-High	3	2	2	1, 2	7

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level-High (Above P-7)	3	2	2	1, 2	7
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7
13. Loss of Flow - Two Loop (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7
14. Steam Generator Water Level-Low-Low	3/loop	2/loop	2/loop	1, 2	7
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7
16. Undervoltage-Reactor Coolant Pumps (Above P-7)	3-1/bus	2	2	1	7
17. Underfrequency-Reactor Coolant Pumps (Above P-7)	3-1/bus	2	2	1	7
18. Turbine Trip (Above P-9)					
a. Emergency Trip Header Low Pressure	3	2	2	1	7
b. Turbine Stop Valve Closure	4	4	4	1	8

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip (Above P-7)	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2 2	1 1	2 2	1, 2 3*, 4*, and 5*	1, 40 39, 40
22. Automatic Trip Logic	2 2	1 1	2 2	1, 2 3*, 4*, and 5*	1 39
23. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2	44
b. Power Range Neutron Flux, P-8	4	2	3	1	44
c. Power Range Neutron Flux, P-9	4	2	3	1	44
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	44
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	44

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TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

(1) Trip function may be manually bypassed in this MODE above P-10.

(2) Trip function may be manually bypassed in this MODE above P-6.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. Less than or equal to 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise reduce thermal power to less than 5% RATED THERMAL POWER within the following 6 hours.

b. Above 5% of RATED THERMAL POWER, operation may continue provided all of the following conditions are satisfied:

    1. The inoperable channel is placed in the tripped condition within 1 hour.

    2. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

    3. Either, THERMAL POWER is restricted to $\leq$75% of RATED THERMAL and the Power Range, Neutron Flux trip setpoint is reduced to <85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.C.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 setpoint.

TABLE 3.3-1 (Continued)

b. Above P-6 but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above P-6 setpoint and suspend positive reactivity operations.

b. Above P-6, operation may continue.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, suspend all operations involving positive reactivity changes and verify Valve 2CHS-91 are closed and secured in position within the next hour.

ACTION 6 - This Action is not used.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. Less than or equal to 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; restore the inoperable channel to operable status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the following 6 hours.

b. Above 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level above P-9, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 9 - This ACTION is not used

TABLE 3.3-1 (Continued)

ACTION 10 - This Action is not used.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 39 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 40 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 1 or ACTION 39 as applicable. Neither breaker shall be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 44 - With less than the Minimum Number of channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux (Below P-10)	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 4.0 seconds*
8. Overpower ΔT	$\leq$ 4.0 seconds*
9. Pressurizer Pressure--Low (Above P-7)	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High (Above P-7)	NOT APPLICABLE
12. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
13. Loss of Flow - Two Loop (Above P-7 and below P-8)	$\leq$ 1.0 seconds
14. Steam Generator Water Level--Low-Low (Loop Stop Valves Open)	$\leq$ 2.0 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps (Above P-7)	$\leq$ 1.5 seconds
17. Underfrequency-Reactor Coolant Pumps (Above P-7)	$\leq$ 0.9 seconds

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*Neutron detectors are exempt from response time testing. Response time shall be measured from detector output or input of first electronic component in channel.

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
18. Turbine Trip (Above P-9)	
A. Emergency Trip Header Low Pressure	NOT APPLICABLE
B. Turbine Stop Valve Closure	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip (Above P-7)	NOT APPLICABLE
21. Reactor Trip Breakers	NOT APPLICABLE
22. Automatic Trip Logic	NOT APPLICABLE
23. Reactor Trip System Interlocks	NOT APPLICABLE

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	S/U(1), R(10)	1, 2, 3*, 4*, 5*
2.	Power Range, Neutron Flux				
	a. High Setpoint	S	D(2), M(3) and Q(6)	M	1, 2
	b. Low Setpoint	S	R	S/U(1)	1(7), 2
3.	Power Range, Neutron Flux, High Positive Rate	N.A.	R	M	1, 2
4.	Power Range, Neutron Flux, High Negative Rate	N.A.	R	M	1, 2
5.	Intermediate Range, Neutron Flux	S	R	S/U(1), M(7)	1, 2, 3*, 4*, 5*
6.	Source Range, Neutron Flux (Below P-10)	S	R	S/U(1), M(8)	2, 3*, 4* and 5*
7.	Overtemperature ΔT	S	R	M	1, 2
8.	Overpower ΔT	S	R	M	1, 2
9.	Pressurizer Pressure-Low (Above P-7)	S	R	M	1, 2
10.	Pressurizer Pressure-High	S	R	M	1, 2

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11. Pressurizer Water Level-High (Above P-7)	S	R	M	1, 2
12. Loss of Flow - Single Loop (Above P-8)	S	R	M	1
13. Loss of Flow - Two Loop (Above P-7 and below P-8)	S	R	See 12 above	1
14. Steam/Generator Water Level-Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage - Reactor Coolant Pumps (Above P-7)	N.A.	R	M	1
17. Underfrequency - Reactor Coolant Pumps (Above P-7)	N.A.	R	M	1
18. Turbine Trip (Above P-9)				
A. Emergency Trip Header Low Pressure	N.A.	R	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	R	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	R	1, 2
20. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	M(5, 11) and S/U(1)	1, 2, 3*, 4*, 5*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
23. Reactor Trip System Interlocks				
A. Intermediate Range Neutron Flux, P-6	N.A.	R	M(9)	1, 2
B. Power Range Neutron Flux, P-8	N.A.	R	M(9)	1
C. Power Range Neutron Flux, P-9	N.A	R	M(9)	1
D. Power Range Neutron Flux, P-10	N.A.	R	M(9)	1, 2
E. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M(9)	1
24. Reactor Trip Bypass Breakers	N.A.	N.A.	M(12),R(13)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial imbalance above 15% of RATED THERMAL POWER. Recalibrate if absolute difference > 3 percent.

(4) - (Not used)

(5) - Each train tested every other month on a STAGGERED TEST BASIS.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-10.

(8) - Below P-6.

(9) - Required only when below Interlock Trip Setpoint.

(10) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) - Local manual shunt trip prior to placing breaker in service.

(13) - Automatic undervoltage trip.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4 adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   EQUATION 2.2-1 $Z + R + S \leq TA$

   where:

   Z = The value for Column Z of Table 3.3-4 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Drift) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each engineered safety feature actuation system instrumentation channel and interlock and the automatic actuation logic with master and slave relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements* during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by interlock operation. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

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*For the automatic actuation logic, the surveillance requirements shall be the application of various simulated input conditions in conjunction with each possible interlock logic state and verification of the required logic output including, as a minimum, a continuity check of output devices. For the actuation relays, the surveillance requirements shall be the energization of each master and slave relay and verification of OPERABILITY of each relay. The test of master relays shall include a continuity check of each associated slave relay. The test of slave relays (to be performed at least once per 92 days in lieu of at least once per 31 days) shall include, as a minimum, a continuity check of associated actuation devices that are not testable.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	SAFETY INJECTION AND FEEDWATER ISOLATION					
	a. Manual Initiation	2	1	2	1, 2, 3, 4	18
	b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13, 36
	c. Containment Pressure-High	3	2	2	1, 2, 3	14
	d. Pressurizer Pressure-Low	3	2	2	1, 2, 3#	14
	e. Steamline Pressure - Low	3/loop	2/loop any loop	2/operating loop	1, 2, 3#	14
1.1	SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RECIRCULATION MODE					
	a. Automatic Actuation Logic Coincident with Safety Injection Signal	2	1	2	1, 2, 3, 4	18
	b. Refueling Water Storage Tank Level-Extreme Low	4	2	3	1, 2, 3, 4	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. CONTAINMENT SPRAY					
a. Manual Initiation	2 sets (2 switches/set)	1 set	2 sets	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Intiation	2 sets (2 switches/set)	1 set	2 sets	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13
3) Containment Pressure--High-High	4	2	3	1, 2, 3, 4	16

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual Initiation					
1. Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3,	41
2. System	2 sets (2 switches/set)	1 set	2 sets	1, 2, 3,	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3,	13
c. Containment Pressure Intermediate-High-High	3	2	2	1, 2, 3	14
d. Steamline Pressure--Low	3/loop	2/loop any loop	2/operating loop	1, 2, 3#	14
e. Steamline Pressure Rate -- High Negative	3/loop	2/loop any loop	2/operating loop	3##	34
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High, P-14	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	42
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. LOSS OF POWER					
a. 4.16kv Emergency Bus					
1. Undervoltage (Trip Feed)	2/4.16kv Bus	2/4.16kv Bus	2/4.16kv Bus	1, 2, 3, 4	33
2. Undervoltage (Start Diesel)	1/4.16kv Bus	1/4.16kv Bus	1/4kv Bus	1, 2, 3, 4	33
b. 4.16kv Emergency Bus (Degraded Voltage)	2/4.16kv Bus	2/Bus	2/Bus	1, 2, 3, 4	34
c. 480 Volt Emergency Bus (Degraded Voltage)	2/480v Bus	2/Bus	2/Bus	1, 2, 3, 4	34
7. AUXILIARY FEEDWATER*					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	42
b. Steam Gen. Water Level--Low-Low					
1. Start Turbine Driven Pump	3/stm. gen.	2/stm. gen. any stm. gen.	2/stm. gen.	1, 2, 3	14
2. Start Motor Driven Pumps	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen.	1, 2, 3	14
c. Undervoltage-RCP (Start Turbine Driven Pump)	(3)-1/bus	2	2	1, 2	14

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*Manual initiation is included in Specification 3.7.1.2

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. AUXILIARY FEEDWATER (Continued)					
d. Safety Injection (Start Motor-Driven Pumps)	See 1 above (all SI initiating functions and requirements)				
e. Turbine Driven Pump Discharge Pressure Low With Steam Valve Open (Start Motor-Driven Pumps)	(2)-1/Train	1	1	1, 2, 3	18
f. Trip of Main Feedwater Pumps (Start Motor Driven Pumps)	1/ pump	2	2	1, 2, 3	18
8. ENGINEERED SAFETY FEATURE INTERLOCKS					
a. Reactor Trip, P-4	2	1	2	1, 2, 3	45
b. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	38
c. Low-Low T_{avg}, P-12	3	2	2	1, 2, 3	38

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip Function automatically bypassed above P-11, and is bypassed below P-11 when Safety Injection on low steam pressure is not manually bypassed.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within six hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to two hours for surveillance testing per Specification 4.3.2.1.1., provided the other channel is operable.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the tripped condition within 1 hour; restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT STANDBY within the following six hours.

b. Above P-11 and P-12, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 15 - (This ACTION is not used)

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the bypass condition; restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT SHUTDOWN within the following 12 hours.

b. Above P-11 or P-12, demonstrate that the Minimum Channels OPERABLE requirement is met within 1 hour; operation may continue with the inoperable channel bypassed and one channel may be bypassed for up to 2 hours for testing per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

ACTION 33 - With the number of OPERABLE Channels one less than the Total Number of Channels, the Emergency Diesel Generator associated with the 4kv Bus shall be declared inoperable and the ACTION Statements for Specifications 3.8.1.1 or 3.8.1.2, as appropriate shall apply.

ACTION 34 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until the performance of the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 36 - The block of the automatic actuation logic introduced by a reset of safety injection shall be removed by resetting (closure) of the reactor trip breakers within one hour of an inadvertent initiation of safety injection providing that all trip input signals have reset due to stable plant conditions. Otherwise, the requirements of ACTION Statement 13 shall have been met.

ACTION 37 - (This ACTION is not used)

ACTION 38 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) (bistable status lights or computer checks) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 41 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 42 - With the number of OPERABLE Channels one less than the minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1 provided the other Channel is OPERABLE.

ACTION 45 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR DRIFT (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. SAFETY INJECTION AND FEEDWATER ISOLATION					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure - High	2.6	0.71	1.67	≤ 1.5 psig	≤ 2.0 psig
d. Pressurizer Pressure - Low	12.5	10.71	1.67	≥ 1845 psig	≥ 1840 psig
e. Steamline Pressure - Low	14.2	1.21	1.67	≥ 500 psig*	≥ 479 psig*
1.1 SAFETY INJECTION TRANSFER FROM INJECTION TO THE RECIRCULATION MODE					
a. Automatic Actuation Logic, Coincident with Safety Injection Signal	N.A.	N.A.	N.A.	N.A.	N.A.
b. Refueling Water Storage Tank Level - Extreme Low	2.8	0.64	1.30	≥ 37 feet 6 inches	≥ 37 feet 4 inches
2. CONTAINMENT SPRAY					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

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*Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR DRIFT (S)	TRIP SETPOINT	ALLOWABLE VALUE
2. CONTAINMENT SPRAY (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure -- High-High	2.9	0.71	1.67	≤ 8.0 psig	≤ 8.7 psig
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3. Safety Injection	See Functional Unit 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3. Containment Pressure-- High-High	2.9	0.71	1.67	≤ 8.0 psig	≤ 8.7 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR DRIFT (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. STEAM LINE ISOLATION					
a. Manual Initiation					
1. Individual	N.A	N.A	N.A	N.A	N.A
2. System	N.A	N.A	N.A	N.A	N.A
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--Intermediate-High-High	2.9	0.71	1.67	≤ 3.0 psig	≤ 3.7 psig
d. Steam Line Pressure - Low	14.2	1.21	1.67	≥ 500 psig*	≥ 479 psig*
e. Steamline Pressure Rate - High Negative	3.0	0.50	0	≤ 100 psi with a time constant ≥ 50 seconds	≤ 121 psi with a time constant ≥ 50 seconds
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level -- High-High, P-14	5.0	2.18	1.67	≤ 75% of narrow range instrument span	≤ 76.8% of narrow range instrument span
c. Safety Injection	See Functional Unit 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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*Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR DRIFT (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. LOSS OF POWER					
a. 4.16 kV Emergency Bus					
1. Undervoltage (Trip Feed)	15.0	1.39	0.0	≥ 75% of nominal Bus Voltage with a 1 ± 0.1 second time delay	≥ 73% of nominal Bus Voltage with a 1 ± 0.1 second time delay
2. Undervoltage (Start Diesel)	15.0	1.39	0.0	≥ 75% of nominal Bus Voltage, 20 cycles ± 2 cycles	≥ 73% of nominal Bus Voltage, 20 cycles ± 2 cycles
b. 4.16 kV Emergency Bus (Degraded Voltage)	15.0	1.39	0.0	≥ 90% of nominal Bus Voltage with a 90 ± 5 second time delay	≥ 88% of nominal Bus Voltage with a 90 ± 5 second time delay
c. 480 Volt Emergency Bus (Degraded Voltage)	15.0	1.39	0.0	≥ 90% of nominal Bus Voltage with a 90 ± 5 second time delay	≥ 88% of nominal Bus Voltage with a 90 ± 5 second time delay
7. AUXILIARY FEEDWATER*					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

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*Manual initiation is included in Specification 3.7.1.2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR DRIFT (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. AUXILIARY FEEDWATER (Continued)					
b. Steam Generator Water Level--Low-Low					
1. Start Turbine Driven Pump	15.5	14.18	1.67	≥ 15.5% of narrow range instrument span	≥ 14.7% of narrow range instrument span
2. Start Motor Driven Pumps	15.5	14.18	1.67	≥ 15.5% of narrow range instrument span	≥ 14.7% of narrow range instrument span
c. Undervoltage - RCP (Start Turbine Driven Pump)	27.7	1.39	0.0	≥ 75% of nominal bus voltage	≥ 73% of nominal bus voltage
d. Safety Injection (Start Motor-Driven Pumps)	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Turbine Driven Pump Discharge Pressure Low with Steam Valve Open (Start Motor-Driven Pumps)	5.0	2.0	0	Discharge pressure ≥ 150 psig with steam inlet valves open	Discharge pressure ≥ 145 psig with steam inlet valves open
f. Trip of Main Feedwater Pumps (Start Motor-Driven Pumps)	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR DRIFT (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. ENGINEERED SAFETY FEATURE INTERLOCKS					
a. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
b. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 2000 psig	≤ 2010 psig
c. Low-Low T_{avg}, P-12	4.0	0.82	0.87	≥ 541°F	≥ 538.5°F

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Vent and Purge Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
b. Containment Quench Spray Pumps	Not Applicable
Containment Quench Spray Valves	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
d. Control Room Ventilation Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0*
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0(1)
d. Containment Isolation-Phase "A"	$\leq$ 61.5(4)/115.5(5)
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Service Water System	$\leq$ 72.5(2)/181.5(3)
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0*/27.0#
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0(1)
d. Containment Isolation-Phase "A"	$\leq$ 61.0(4)/115.0 (5)

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low (Continued)	
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Service Water System	≤ 72.0(2)/181.0(3)
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 37.0##/27.0#
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0(1)
d. Containment Isolation-Phase "A"	≤ 61.0(4)/115.0(5)
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Service Water System	≤ 72.0(2)/181.0(3)
g. Steam Line Isolation	≤ 7.0
5. Containment Pressure--High-High	
a. Containment Quench Spray	≤ 85.5(5)
b. Containment Isolation-Phase "B"	Not Applicable
c. Control Room Ventilation Isolation	≤ 22.0(4)/77.0(5)
6. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 7.0(1)
7. Containment Pressure--Intermediate High-High	
a. Steam Line Isolation	≤ 7.0
8. Steamline Pressure Rate--High Negative	
a. Steamline Isolation	≤ 7.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
9. Loss of Power	
a. 4.16kv Emergency Bus Undervoltage (Loss of Voltage) (Trip Feeder)	1 ± 0.1 sec.
b. 4.16kv and 480v Emergency Bus Undervoltage (Degraded Voltage)	90 ± 5 sec.
10. (Intentionally blank)	
11. Steam Generator Water Level-Low-Low	
a. Motor-driven Auxiliary Feedwater Pump**	≤ 60.0
b. Turbine-driven Auxiliary Feedwater Pump***	≤ 60.0
12. Undervoltage RCP	
a. Turbine-driven Auxiliary Feedwater Pump	≤ 60.0
13. Trip of Main Feedwater Pumps	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60.0
14. Turbine Driven Auxiliary Feedwater Pump Discharge Pressure Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60.0
15. Control Room High Radiation	
a. Control Room Ventilation Isolation	≤ 180(6)

**on 2/3 in 2/3 Steam Generators
***on 2/3 any Steam Generator

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps and Low Head Safety Injection pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is not included.

# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.

## Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.

(1) Feedwater system overall response time shall include verification of valve stroke times applicable to the feedwater valves shown for penetrations 76, 77 and 78 shown in Table 3.6-1.

(2) Diesel generator starting and sequence loading delays included. Response time limit includes attainment of discharge pressure for service water pumps.

(3) Diesel generator starting and sequence loading delays not included. Response time limit only includes opening of valves to establish the flowpath to the diesel coolers.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes operation of valves/dampers.

(5) Diesel generator starting, and sequence loading delays included. Response time limit includes operation of valves/dampers.

(6) Diesel generator starting and sequence loading delays not included. Response time limit includes operation of dampers.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTI[illegible]ND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M	1, 2, 3
e. Steam Line Pressure--Low	S	R	M	1, 2, 3
1.1 SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RECIRCULATION MODE				
a. Automatic Actuation Logic, Coincident with Safety Injection Signal	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Refueling Water Storage Tank Level-Extreme Low	S	R	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation and Logic Actuation Relays	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M	1, 2, 3
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(2)	1, 2, 3, 4
3. Safety Injection	See Functional Unit 1. above for all Safety Injection Surveillance Requirements.			
b. Phase "B" Isolation				
1. Manual Initation	N.A.	N.A.	M(1)	1, 2, 3, 4
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(2)	1, 2, 3, 4
3. Containment Pressure--High-High	S	R	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual Initiation				
1. Individual	N.A.	N.A.	M(1)	1, 2, 3
2. System	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--Intermediate-High-High	S	R	M	1, 2, 3
d. Steam Line Pressure--Low	S	R	M	1, 2, 3
e. Steamline Pressure Rate-High Negative	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(2)	1, 2, 3
b. Steam Generator Water Level--High-High, P-14	S	R	M	1, 2, 3
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements.			

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. LOSS OF POWER				
a. 4.16kv Emergency Bus	N.A.	R	M	1, 2, 3, 4
1. Undervoltage (Trip Feed)				
2. Undervoltage (Start Diesel)	N.A.	R	M	1, 2, 3, 4
b. 4.16kv Emergency Bus (Degraded Voltage)	N.A.	R	M	1, 2, 3, 4
c. 480 Volt Emergency Bus (Degraded Voltage)	N.A.	R	M	1, 2, 3, 4
7. AUXILIARY FEEDWATER*				
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(2)	1, 2, 3
b. Steam Generator Water Level--Low-Low				
1. Start Turbine Driven Pump	S	R	M	1, 2, 3
2. Start Motor Driven Pumps	S	R	M	1, 2, 3
c. Undervoltage - RCP (Start Turbine Driven Pump)	S	R	M	1, 2

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*Manual initiation is included in Specification 4.7.1.2.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. AUXILIARY FEEDWATER (continued)				
d. Safety Injection (Start Motor-Driven Pumps)	See 1 above (all SI surveillance requirements)			
e. Turbine-driven Pump Discharge Pressure Low With Steam Valve Open (Start Motor Driven Pumps)	N.A.	R	R	1, 2, 3
f. Trip of Main Feedwater Pumps (Start Motor-Driven Pumps)	N.A.	N.A.	R	1, 2, 3
8. ENGINEERED SAFETY FEATURE INTERLOCKS				
a. Reactor Trip, P-4	N.A.	N.A.	R	1, 2, 3
b. Pressurizer Pressure, P-11	N.A.	R	M	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) Each train or logic channel shall be tested at least every other 31 days.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT#	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area (2RMF-RQ202)	1	*	$\leq$ 75.8 mR/hr	10^{-1} to 10^{4} mR/hr	19
b. Containment Area (2RMR-RQ206 & 207)	2	1, 2, 3 & 4	$\leq 3.29 \times 10^{3}$ R/hr	1 to 10^{7} R/hr	36
c. Control Room Area (2RMC-RQ201 & 202)	2	1, 2, 3, 4, 5## & 6##	$\leq$ 0.476 mR/hr	10^{-2} to 10^{3} mR/hr	46, 47
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity (Xe-133) RCS Leakage Detection (2RMR-RQ303B)	1	1, 2, 3 & 4	N/A	10^{-6} to 10^{-1} μCi/cc	20
ii. Particulate Activity (I-131) RCS Leakage Detection (2RMR-RQ303A)	1	1, 2, 3 & 4	N/A	10^{-10} to 10^{-5} μCi/cc	20
b. Fuel Building Vent					
i. Gaseous Activity (Xe-133) (2RMF-RQ301B)	1	**	$\leq 7.82 \times 10^{-6}$ μCi/cc	10^{-6} to 10^{-1} μCi/cc	21

---

*With fuel in the storage pool or building

**With Irradiated fuel in the storage pool

#Above background

##During movement of irradiated fuel

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT#	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
ii. Particulate (I-131) (2RMF-RQ301A)	1	**	$\leq 6.70\times10^{-9}$ µCi/cc	10^{-10} to 10^{-5} µCi/cc	21
c. Noble Gas and Effluent Monitors					
i. Supplementary Leak Collection and Release System					
1) Mid Range Noble Gas (Xe-133)(2HVS-RQ109C)	1	1, 2, 3 & 4	N.A.	10^{-4} to 10^{2} µCi/cc	36
2) High Range Noble Gas (Xe-133)(2HVS-RQ109D)	1	1, 2, 3 & 4	N.A.	10^{-1} to 10^{5} µCi/cc	36
ii. Containment Purge Exhaust (Xe-133)(2HVR-RQ104A & B)	1	6	$\leq$ 3 x background	10^{-6} to 10^{-1} µCi/cc	22
iii. Main Steam Discharge (Kr-88)(2MSS-RQ101A,B & C)	1/SG	1, 2, 3 & 4	$\leq 3.9\times10^{-2}$ µCi/cc	10^{-2} to 10^{3} µCi/cc	36

#Above background

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specifications 3.9.12 and 3.9.13.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 36 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 43 - With the number of OPERABLE channels less than required by the Minimum channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Return the channel to OPERABLE status within 30 days or explain in the next Semi-Annual Effluent Release Report why the inoperability was not covered in a timely manner.

ACTION 46 - With the number of OPERABLE channels one less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel to OPERABLE status within 7 days or close the control room series normal air intake and exhaust isolation dampers.

ACTION 47 - With no OPERABLE channels either restore one inoperable channel to OPERABLE status within 1 hour or close the control room series normal air intake and exhaust isolation dampers.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area (2RMF-RQ202)	S	R	M	*
b. Containment Area (2RMR-RQ206 & 207)	S	R	M	1, 2, 3, 4
c. Control Room Area (2RMC-RQ201 & 202)	S	R	M	1, 2, 3, 4, 5## & 6##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (2RMR-RQ303B)	S	R	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection (2RMR-RQ303A)	S	R	M	1, 2, 3 & 4
b. Fuel Building Vent				
i. Gaseous Activity (2RMF-RQ301B)	S	R	M	**
ii. Particulate Activity (2RMF-RQ301A)	S	R	M	**

---

*With fuel in the storage pool or building
**With irradiated fuel in the storage pool
##During movement of irradiated fuel

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TABLE 4.3-3 (Continued)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
i. Supplementary Leak Collection and Release System (2HVS-RQ109C & D)	S	R	M	1, 2, 3 & 4
ii. Containment Purge Exhaust (2HVR-RQ104A & B)	S	R	M	6
iii. Main Steam Discharge (2MSS-RQ101A, B & C)	S	R	M	1, 2, 3 & 4

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

A. Recalibration of the axial flux offset detection system,

B. Monitoring the QUADRANT POWER TILT RATIO, or

C. Measurement of $F_{\Delta H}^{N}$ and F_Q (Z).

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore movable detection system shall be demonstrated OPERABLE by normalizing each detector output to be used within 24 hours prior to its use when required for:

a. Recalibration of the excore axial flux offset detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_{\Delta H}^{N}$ and F_Q (Z).

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE seismic monitoring instruments less than required by Table 3.3-7, restore the inoperable instrument(s) to OPERABLE status within 30 days.

b. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 A seismic event greater than or equal to 0.01g shall be reported to the Commission within 1 hour. Each of the above seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 30 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

TABLE 3.3-7
SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. TRIAXIAL TIME-HISTORY ACCELEROGRAPHS (1)(2)(4)		
a. Containment Mat el. 692'-11"	± 1 g	1
b. Containment Operating Floor el. 767'-10"	± 1 g	1
c. Switchyard	± 1 g	1
d. Containment Building - Steam Generator Support Cubicle No. 1 el. 718'6"	± 1 g	1
e. Aux. Building - at center of Mat, el 710'6"	± 1 g	1
f. Aux. Building - at base of 480 volt MCC (MCC-2-E03), el. 755'6"	± 1 g	1
2. TRIAXIAL PEAK ACCELEROGRAPHS		
a. Containment Bulding - RHS heat exchanger (2RHS-E21A) el. 715'-6"	± 2 g	1
b. Containment Building - Six Inch SI Pipe (2SIS-006-269-1(A)) el 741'-5"	± 2 g	1
c. Aux. Building - MCC-2-E03 el. 755'-6"	± 5 g	1
3. TRIAXIAL SEISMIC SWITCH (3)		
a. Containment mat	N/A	1
4. RESPONSE SPECTRUM ANALYZER		
a. Control Room	N/A	1

NOTES

(1) Units a, b, c are wired to accelerograph recorders in the Control Room. Units d, e, and F are self contained units

(2) Each accelerograph trigger setpoint is set at 0.01g.

(3) Triaxial seismic switch setpoints:
- horizontal sensor: 0.049g
- vertical sensor: 0.037g

(4) Triaxial time-history accelerograph - Units a and c are input directly to the response spectrum analyzer in the Control Room.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. TRIAXIAL TIME-HISTORY ACCELEROGRAPHS			
a. Containment Mat. el. 692'-11"	M*	R	SA
b. Containment Operating floor el. 767'-10"	M*	R	SA
c. Switchyard	M*	R	SA
d. Containment Building - Steam Generator Support Cubicle No. 1 el. 718'6"	N/A	R	N/A
e. Aux. Building - at center of Mat, el 710'6"	N/A	R	N/A
f. Aux. Building - at base of 480 volt MCC (MCC-2-E03), el. 755'6"	N/A	R	N/A
2. TRIAXIAL PEAK ACCELEROGRAPHS			
a. Containment Building - RHS heat exchanger (2RHS-E21A) el. 715'-6"	N/A	R	N/A
b. Containment Building -Six inch SI pipe (2SIS-006-269-1(A)) el. 741'-5"	N/A	R	N/A
c. Aux. Building - MCC-2E03 el 755'-6"	N/A	R	N/A
3. TRIAXIAL SEISMIC SWITCHES			
a. Containment mat	N/A	N/A	R
4. RESPONSE SPECTRUM ANALYZER			
a. Control Room	N/A	R	N/A

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*Except seismic trigger

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METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE meteorological monitoring channels less than required by Table 3.3-8, suspend all release of gaseous radioactive material from the radwaste gas decay tanks until the inoperable channel(s) is restored to OPERABLE status.

b. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	INSTRUMENT MINIMUM ACCURACY	MINIMUM OPERABLE
1. WIND SPEED			
a. Nominal Elev. 500'		± 0.5 mph*	1
b. Nominal Elev. 150'		± 0.5 mph*	1
c. Nominal Elev. 35'		± 0.5 mph*	1
2. WIND DIRECTION			
a. Nominal Elev. 500'		± 5°	1
b. Nominal Elev. 150'		± 5°	1
c. Nominal Elev. 35'		± 5°	1
3. AIR TEMPERATURE ΔT			
a. ΔT Elev. 500'-35'		± 0.1°C	1
b. ΔT Elev. 150'-35'		± 0.1°C	1

---

*Starting speed of anemometer shall be < 1 mph.

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 500'	D	SA
b. Nominal Elev. 150'	D	SA
c. Nominal Elev. 35'	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 500'	D	SA
b. Nominal Elev. 150'	D	SA
c. Nominal Elev. 35'	D	SA
3. AIR TEMPERATURE ΔT		
a. ΔT Elev. 500'-35'	D	SA
b. ΔT Elev. 150'-35'	D	SA

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either:

a. Restore the inoperable channel to OPERABLE status within 7 days, or

b. Be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN PANEL MONITORING INSTRUMENTATION

	INSTRUMENTS*	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	Intermediate Range Nuclear Flux	10^{-11} to 10^{-3} amps	1
2.	Intermediate Range Startup Rate	-0.5 to +5.0 DPM	1
3.	Source Range Nuclear Flux	10^{0} to 10^{6} CPS	1
4.	Source Range Startup Rate	-0.5 to +5 DPM	1
5.	Reactor Coolant Temperature - Hot Leg	0 - 700°F	1
6.	Reactor Coolant Temperature - Cold Leg	0 - 700°F	1
7.	Pressurizer Pressure	1700 to 2500 psig	1
8.	Pressurizer Level	0 - 100%	1
9.	Steam Generator Pressure	0 - 1200 psig	1/steam generator
10.	Steam Generator Level	0 to 100%	1/steam generator
11.	RHR Temperature - HX Outlet	50 - 400°F	1
12.	Auxiliary Feedwater Flow Rate	0 - 400 GPM	1/steam generator

---

*Emergency Shutdown Panel

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENTS*	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Intermediate Range Nuclear Flux	M	N.A.
2.	Intermediate Range Startup Rate	M	N.A.
3.	Source Range Nuclear Flux (2)	M(4)	N.A.
4.	Source Range Startup Rate (2)	M(4)	N.A.
5.	Reactor Coolant Temperature - Hot Leg	M	R
6.	Reactor Coolant Temperature - Cold Leg	M	R
7.	Pressurizer Pressure	M	R
8.	Pressurizer Level	M	R
9.	Steam Generator Pressure	M	R
10.	Steam Generator Level	M	R
11.	RHR Temperature - HX Outlet (3)	M	R
12.	Auxiliary Feedwater Flow Rate	S/U(1)	R

---

*Emergency Shutdown Panel

(1) Channel check to be performed in conjunction with Surveillance Requirement 4.7.1.2.b following an extended plant outage.

(2) Operability required in accordance with Specification 3.3.1.1.

(3) Operability required in accordance with Specification 3.4.1.3.

(4) Below P-6.

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3/4.3.3.6 (This Specification number is not used.)

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Three independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6.

ACTION:

a. With one chlorine detection system inoperable, operation may continue provided the inoperable detector is placed in the tripped condition within 1 hour.

b. With two chlorine detection systems inoperable, restore one of the inoperable detection systems to OPERABLE status within 7 days, or within the next 6 hours, isolate outside air inlet to the control room.

c. With no chlorine detection system OPERABLE, within 1 hour isolate outside air inlet to the control room.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTION TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3.11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3.11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours except for the PORV(s) which may be isolated in accordance with Specification 3.4.11.a.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE Reactor Vessel Level Indication System channels less than the required number of channels or the Minimum Channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status as per ACTION a or b above as applicable if repair is not feasible, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days that provides action taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status. This ACTION statement applies to the first fuel cycle only.

d. With the number of OPERABLE Reactor Coolant System Subcooling Margin Monitor instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in a least HOT SHUTDOWN within the next 12 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1.	Pressurizer Water Level	3	2	a, b
2.	Auxiliary Feedwater Flow Rate	2 per steam generator	1 per steam generator	a, b
3.	Reactor Coolant System Subcooling Margin Monitor	2	1	d
4.	PORV Limit Switch Position Indicator	1/valve	0/valve	a, b
5.	PORV Block Valve Limit Switch Position Indicator	1/valve	0/valve	a, b
6.	Safety Valve Position Indicator	1/valve	0/valve	a, b
7.	Safety Valve Temperature Detector	1/valve	0/valve	a, b
8.	Containment Sump Wide Range Water Level	2	1	a, b
9.	Containment Wide-Range Pressure	2	1	a, b
10.	Reactor Vessel Level Indication System	2	1	a, b, c*
11.	Core Exit Thermocouples	4/core quadrant	2/core quadrant	a, b

---

*Only ACTION statement c is applicable during the first fuel cycle. ACTION statements a and b are applicable thereafter.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Pressurizer Water Level	M	R
2.	Auxiliary Feedwater Flow Rate	S/U*	R
3.	Reactor Coolant System Subcooling Margin Monitor	M	R
4.	PORV Limit Switch Position Indicator	M	R
5.	PORV Block Valve Limit Switch Position Indicator	M	R
6.	Safety Valve Position Indicator	M	R
7.	Safety Valve Temperature Detector	M	R
8.	Containment Sump Wide-Range Water Level	M	R
9.	Containment Wide-Range Pressure	N/A	R
10.	Reactor Vessel Level Indication System	M	R
11.	Core Exit Thermocouples	M	R

---

*Channel check to be performed in conjunction with Surveillance Requirement 4.7.1.2.b following an extended plant outage.

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of the radiation monitoring channels shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: During releases through the flow path.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or correct the alarm/trip setpoint.

b. With one or more radioactive liquid effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.3-12 or conservatively reduce the alarm setpoint. Exert a best effort to return the channel to operable status within 30 days, and if unsuccessful, explain in the next Semi-Annual Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-12.

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TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Radioactivity Monitor Providing Alarm and Automatic Termination of Release		
a) Liquid Waste Process Effluent Monitor (2SGC-RQ100)	1	23
2. Gross Radioactivity Monitors Providing Alarm But Not Providing Termination of Release		
None		
3. Flow Rate Measurement Devices		
a) Liquid Radwaste Effluent (2SGC-FS100)	1	25
b) Cooling Tower Blowdown Line (2CWS-FT101)	1	25
4. Tank Level Indicating Devices (For tanks outside plant buildings)		
None		

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may be resumed provided that prior to initiating a release;

1. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and;
2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 24 - (This ACTION is not used)

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

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TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Radioactivity Monitor Providing Alarm and Automatic Termination of Release				
a) Liquid Waste Process Effluent (2SGC-RQ100)	D	P(5)	R(2)(3)	Q(1)
2. Gross Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
None				
3. Flow Rate Measurement Devices				
a) Liquid Radwaste Effluent (2SGC-FS100)	D(4)	N/A	R	Q
b) Cooling Tower Blowdown Line (2CWS-FT101)	D(4)	N/A	R	Q
4. Tank Level Indicating Devices (For tanks outside plant buildings)				
None				

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TABLE 4.3-12 (Continued)

TABLE NOTATION (Continued)

(1) - The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and Control Room Alarm Annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint.

(2) - The CHANNEL CALIBRATION shall also demonstrate that Control Room alarm annunciation occurs if either of the following conditions exist:

1. Downscale failure.
2. Instrument controls are not set in operate mode.

(3) - The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per eighteen months.

(4) - CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

(5) - A source check may be performed utilizing the installed means or flashing the detector with a portable source to obtain an upscale increase in the existing count rate to verify channel response.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of 3.11.2.1 are not exceeded. The alarm/trip setpoints of the radiation monitoring channels shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: During releases through the flow path.

ACTION:

a. With a radioactive gaseous process or effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.11.2.1 are met, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or correct the alarm/trip setpoint.

b. With one or more radioactive gaseous effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.3-13 or conservatively reduce the alarm setpoint. Exert a best effort to return the channel to operable status within 30 days, and if unsuccessful, explain in the next Semi-Annual Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-13.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	PARAMETER	ACTION
1. Gaseous Waste/Process Vent System (RM-GW-108A & B) (BVPS Unit 1)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	27, 30 ***
b. Particulate Activity Monitor	1	*		32
c. System Effluent Flow Rate Measuring Device (FR-GW-108)	1	*	System Flow Rate Measurement	28
d. Sampler Flow Rate Measuring Device	1	*	Sampler Flow Rate Measurement	28
2. Ventilation System (2HVS-RQ101 A & B)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	29, 30 ***
b. Particulate Activity Monitor	1	*		32
c. Process Flow Rate Monitor	1	*	Process Flow Rate Measurement	28
d. Sampler Flow Rate Monitor	1	*	Sampler Flow Rate Measurement	28

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*During releases via this pathway.
***During purging of Reactor Containment via this pathway.

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TABLE 3.3-13 (Continued)

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	PARAMETER	ACTION
3. Elevated Release (2HVS-RQ109A & B)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	29, 30 ***
b. Particulate Activity Monitor	1	*		32
c. Process Flow Rate Monitor	1	*	Process Flow Rate Measurement	28
d. Sampler Flow Rate Monitor Device	1	*	Sampler Flow Rate Measurement	28
4. Decontamination Building Vent (2RMQ-RQ301A & B)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	29
b. Particulate Activity Monitor	1	*		32
c. Process Flow Rate Monitor	1	*	Process Flow Rate Measurement	28
d. Sampler Flow Rate Monitor	1	*	Sampler Flow Rate Measurement	28

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*During release via this pathway.

***During purging of reactor containment via this pathway

TABLE 3.3-13 (Continued)

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	PARAMETER	ACTION
5. Condensate Polishing Building Vent (2HVL-RQ112A & B)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	29
b. Particulate Activity Monitor	1	*		32
c. Process Flow Rate Monitor	1	*	Process Flow Rate Measurement	28
d. Sampler Flow Rate Monitor	1	*	Sampler Flow Rate Measurement	28
6. Waste Gas Storage Vault (2RMQ-RQ303A & B)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	29
b. Particulate Activity Monitor	1	*		32
c. Process Flow Rate Monitor	1	*	Process Flow Rate Measurement	28
d. Sampler Flow Rate Monitor	1	*	Sampler Flow Rate Measurement	28
7. Gaseous Waste System Surge Tank Discharge				
a. Oxygen Monitor (2GWS-OA100A & B)	2	**	Oxygen	31

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*During release via this pathway.
**During waste gas decay tank filling operation.

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TABLE 3.3-13 (Continued)

ACTION STATEMENTS

ACTION 27 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank may be released to the environment provided that prior to initiating the release:

1. At least two independent samples of the tank's content are analyzed, and at least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup, or
2. Initiate continuous monitoring with a comparable alternate monitoring channel. Surveillance requirements applicable to the inoperable channel shall apply to the comparable alternate monitoring channel when used to satisfy this technical specification requirement.

Otherwise, suspend releases of radioactive effluents via this pathway.

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided:

1. Grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours, or
2. Initiate continuous monitoring with a comparable alternate monitoring channel. Surveillance requirements applicable to the inoperable channel shall apply to the comparable alternate monitoring channel when used to satisfy this technical specification requirement.

ACTION 30 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, immediately suspend PURGING of Reactor Containment via this pathway.

ACTION 31 - With the number of channels OPERABLE one less than required by the MINIMUM Channels OPERABLE requirement, operation of this system may continue provided grab samples are obtained every 4 hours and analyzed within the following 4 hours during additions to a tank.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 or sampled and analyzed once every 8 hours.

ACTION 35 - (This ACTION is not used)

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gaseous Waste/Process Vent System (RM-GW-108A & B) [BVPS Unit 1]				
a. Noble Gas Activity Monitor	P	P(5)	R(3)	Q(1)
b. Particulate Activity Monitor	W	N/A	N/A	N/A
c. System Effluent Flow Rate Measuring Device (FR-GW-108)	P	N/A	R	Q
d. Sampler Flow Rate Measuring Device	D*	N/A	R	Q
2. Ventilation System (2HVS-RQ101A & B)				
a. Noble Gas Activity Monitor	D	M(5), P(5)***	R(3)(6)	Q(2)
b. Particulate Activity Monitor	W	N/A	N/A	N/A
c. Process Flow Rate Monitor	D	N/A	R	Q
d. Sampler Flow Rate	D	N/A	R	Q

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TABLE 4.3-13 (Continued)

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
3. Elevated Release (2HVS-RQ109A & B)				
a. Noble Gas Activity Monitor	D	M(5), P(5)***	R(3)(6)	Q(2)
b. Particulate Activity Monitor	W	N/A	N/A	N/A
c. Process Flow Rate Monitor	D	N/A	R	Q
d. Sampler Flow Rate Monitor	D	N/A	R	Q
4. Decontamination Building Vent (2RMQ-RQ301A & B)				
a. Noble Gas Activity Monitor	D	M(5)	R(3)(6)	Q(2)
b. Particulate Activity Monitor	W	N/A	N/A	N/A
c. Process Effluent Flow Rate Monitor	D	N/A	R	Q
d. Sampler Flow Rate Monitor	D	N/A	R	Q
5. Condensate Polishing Building Vent (2HVL-RQ112A & B)				
a. Noble Gas Activity Monitor	D	M(5)	R(3)(6)	Q(2)
b. Particulate Activity Monitor	W	N/A	N/A	N/A
c. Process Flow Rate Monitor	D	N/A	R	Q
d. Sampler Flow Rate Monitor	D	N/A	R	Q

TABLE 4.3-13 (Continued)

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
6. Waste Gas Storage Vault (2RMQ-RQ303A & B)				
a. Noble Gas Activity Monitor	D	M(5)	R(3)(6)	Q(2)
b. Particulate Activity Monitor	W	N/A	N/A	N/A
c. Process Flow Rate Monitor	D	N/A	R	Q
d. Sampler Flow Rate Monitor	D	N/A	R	Q
7. Gaseous Waste System Surge Tank Discharge				
a. Oxygen Monitor (2GWS-OA100A & B)	D	N/A	Q(4)	M

Table 4.3-13 (Continued)

TABLE NOTATION

* During releases via this pathway

*** During purging of Reactor Containment via this pathway.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and Control Room Alarm Annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Downscale failure.

c. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that Control Room Alarm Annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint.

(3) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per eighteen months. This can normally be accomplished during refueling outages.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and

2. Four volume percent oxygen, balance nitrogen

(5) A source check may be performed utilizing the installed means or flashing the detector with a portable source to obtain an upscale increase in the existing count rate to verify channel response.

(6) The CHANNEL CALIBRATION shall also demonstrate that Control Room Alarm Annunciation occurs if either of the following conditions exist:

1. Downscale failure.

2. Instrument controls are not set in operate mode.

TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 31 days by cycling each of the following valves through at least one complete cycle from the running position:

   1) Four high pressure turbine stop valves,
   2) Four high pressure turbine governor valves,
   3) Four high pressure turbine reheat stop valves, and
   4) Four low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position,

c. At least once per 18 months by performing a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks, and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

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*Specification not applicable with all main steam isolation valves and associated bypass valves in the closed position and all other steam flow paths to the turbine isolated.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

NORMAL OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 All reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.1.2 The power to each of the Reactor Coolant System loop stop valves shall be verified to be removed at least once per 31 days during operation in MODES 1 and 2.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two reactor coolant loops and associated steam generators and reactor coolant pumps shall be in operation* when the rod control system is capable of control bank rod withdrawal.

b. At least two reactor coolant loops and associated steam generators and reactor coolant pumps shall be OPERABLE and one reactor coolant loop shall be in operation* when the rod control system is incapable of control bank rod withdrawal.

APPLICABILITY: MODE 3**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than two reactor coolant loops in operation, immediately deenergize all control rod drive mechanisms, or align the rod control system so that it is incapable of control bank rod withdrawal.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 With the rod control system capable of rod withdrawal, at least two cooling loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.2 With the rod control system incapable of rod withdrawal, at least two cooling loops, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.3 With the rod control system incapable of rod withdrawal, at least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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*All reactor coolant pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature. This does not preclude natural circulation cooldown under abnormal cooldown conditions.

**See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE.

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump,
4. Residual Heat Removal Pump (A) and the (A) RHR heat exchanger,**
5. Residual Heat Removal Pump (B) and the (B) RHR heat exchanger.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation. Refer to Specification 3.4.1.6 for additional limitations.

**The normal or emergency power source may be inoperable in MODE 5.

***All reactor coolant pumps and Residual Heat Removal pumps may be deenergized for up to 1 hour provided: 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5, and by verifying that each residual heat removal pump develops a differential pressure of ≥ 126 psid on recirculation flow.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level greater than or equal to 15.5 percent narrow range at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

3.4.1.4 (This specification number is not used.)

3.4.1.5 (This specification number is not used.)

REACTOR COOLANT PUMP-STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 An idle reactor coolant pump in a non-isolated loop shall not be started, unless the secondary water temperature* of each steam generator is less than 50°F above each of the inservice RCS cold leg temperatures.

APPLICABILITY: When the temperature of one or more of the non-isolated loop cold legs is $\leq$ 350°F.

ACTION:

With the temperature of the steam generator in the loop associated with the reactor coolant pump being started greater than 50°F above the cold leg temperature of the other non-isolated loops, suspend the startup of the reactor coolant pump.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The secondary water temperature of the non-isolated steam generators shall be determined within 10 minutes prior to starting a reactor coolant pump.

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*The secondary water temperature is to be verified by direct measurement of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping.

3/4.4.2 SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting* of 2485 psig ± 1 percent.

APPLICABILITY: MODES 4 and 5

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional requirements other than those required by Specification 4.0.5.

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*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.3 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional requirements other than those required by Specification 4.0.5.

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*The lift setting shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 150 kW of pressurizer heaters and with a steam bubble.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With the pressurizer inoperable due to less than 150 kW of heaters supplied by an emergency bus, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in the HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by energizing the heaters supplied by the emergency bus.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with the method prescribed in Artcle 8 - "Eddy Current Examination of Tubular Products," as contained in ASME Boiler and Pressure Vessel Code, Section V - "Non-destructive Examination," and referenced in ASME Boiler and Pressure Vessel Code - Appendix IV of the 1980 Edition through Winter 1980 Addenda of Section XI - "Inservice Inspection of Nuclear Power Plant Components." The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

    1) All nonplugged tubes that previously had detectable wall penetrations (> than 20%),

    2) Tubes in those areas where experience has indicated potential problems.

SURVEILLANCE REQUIREMENTS (Continued)

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1 percent of the total tubes inspected are defective, or between 5 percent and 10 percent of the total tubes inspected are degraded tubes.
C-3	More than 10 percent of the total tubes inspected are degraded tubes or more than 1 percent of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10 percent) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies.

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under All Volatile Treatment (AVT) conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be increased at least once per 20 months. The increase in the inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

SURVEILLANCE REQUIREMENTS (Continued)

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions.

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections ≥ 20 percent of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40 percent of the nominal tube wall thickness.

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7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Three	Three
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One[1]	One[2]

Table Notation

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9 % of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instruction described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

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1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective Tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S.G.'s are C-1	None	N/A	N/A
			Some S.G.'s C-2 but no add'l S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Add'l S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N/A	N/A

$S = \frac{9}{n} \%$ Where n is the number of steam generators inspected during an inspection.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment sump discharge flow measurement system or narrow range level instrument, and

c Containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above required radioactivity monitoring leakage detection systems inoperable, operations may continue for up to 30 days provided:

   1. The other two above required leakage detection systems are OPERABLE, and

   2. Appropriate grab samples are obtained and analyzed at least once per 24 hours:

   otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment sump discharge flow measurement system and narrow range level instrument inoperable, restore at least one inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable in MODES 1, 2, and 3.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

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SURVEILLANCE REQUIREMENTS (Continued)

b. Containment sump discharge flow measurement system-performance of CHANNEL CALIBRATION at least once per 18 months.

c. Logging the narrow range level indication every 12 hours

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 28 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate and gaseous radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve full open.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

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e. Monitoring the reactor head flange leakoff temperature at least once per 24 hours.

REACTOR COOLANT SYSTEM

PRESSURE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor Coolant System pressure isolation valves as shown in Table 4.4-3 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

1. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit stated in Table 4.4-3, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of a closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.*

2. The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.

SURVEILLANCE REQUIREMENTS

4.4.6.3.1 Leakage testing** of each valve listed in Table 4.4-3 shall be accomplished prior to entering MODE 2 after every time the plant is placed in the COLD SHUTDOWN condition for refueling and prior to returning the valve to service after each maintenance, repair or replacement work is performed; and

4.4.6.3.2 Additional leakage testing of each valve identified by note (d) listed in Table 4.4-3 shall be accomplished prior to entering MODE 2 after each time the plant is placed in COLD SHUTDOWN for 72 hours if testing has not been accomplished in the preceding 9 months.

---

* Motor operated valves shall be placed in the closed position and power supplies de-energized.

** To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating compliance within the valve leakage criteria.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Leakage Rates(a) Allowable/Maximum
Loop 21, Cold leg, LHSI	2SIS-107	$\leq$ 3.0/$\leq$ 5.0 gpm(b)(d)
Loop 22, Cold leg, LHSI	2SIS-108	$\leq$ 3.0/$\leq$ 5.0 gpm(b)(d)
Loop 23, Cold leg, LHSI	2SIS-109	$\leq$ 3.0/$\leq$ 5.0 gpm(b)(d)
Common, Cold leg, LHSI	2SIS-132	$\leq$ 5.0/$\leq$ 5.0 gpm(d)
	2SIS-133	$\leq$ 5.0/$\leq$ 5.0 gpm(d)
Loop 22, Hot leg, LHSI	2SIS-128	$\leq$ 3.0/$\leq$ 5.0 gpm(b)
Loop 23, Hot leg, LHSI	2SIS-129	$\leq$ 3.0/$\leq$ 5.0 gpm(b)
Common, Hot leg, LHSI	2SIS-130	$\leq$ 5.0/$\leq$ 5.0 gpm
Loop 21, Cold leg, SIACC	2SIS-151	$\leq$ 5.0/$\leq$ 5.0 gpm(b)
	2SIS-148	$\leq$ 5.0/$\leq$ 5.0 gpm
Loop 22, Cold leg, SIACC	2SIS-145	$\leq$ 5.0/$\leq$ 5.0 gpm(b)(d)
	2SIS-147	$\leq$ 5.0/$\leq$ 5.0 gpm
Loop 23, Cold leg, SIACC	2SIS-141	$\leq$ 5.0/$\leq$ 5.0 gpm(b)(d)
	2SIS-142	$\leq$ 5.0/$\leq$ 5.0 gpm
Loop 21, Hot leg, RHS-A	2RHS-MOV702A	$\leq$ 5.0/$\leq$ 5.0 gpm(b)
	2RHS-MOV701A	$\leq$ 5.0/$\leq$ 5.0 gpm(b)
Loop 22, Cold leg	2RHS-MOV720A	$\leq$ 5.0/$\leq$ 5.0 gpm(b)(c)
Loop 21, Hot leg, RHS-B	2RHS-MOV702B	$\leq$ 5.0/$\leq$ 5.0 gpm(b)
	2RHS-MOV701B	$\leq$ 5.0/$\leq$ 5.0 gpm(b)
Loop 23, Cold leg	2RHS-MOV720B	$\leq$ 5.0/$\leq$ 5.0 gpm(b)(c)

(a) At function pressure:

1. Leakage rates less than or equal to 0.5 gpm/inch diameter are acceptable.

2. Leakage rates greater than 0.5 gpm/inch diameter but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50 percent or greater.

3. Leakage rates greater than 0.5 gpm/inch diameter but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50 percent or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

5. Observed leakage rates shall be adjusted to the function maximum pressure in accordance with ASME XI IWV 3423.

(b) Minimum test differential pressures shall not be less than 150 psid.

(c) Leakage rate continuously monitored during plant operation, no other leakage rate testing required. Leakage rate acceptance criteria shall be as stated in (a) and (b) above and shall be recorded at intervals as noted in paragraph 4.4.6.3.1 as a minimum.

(d) Both surveillances 4.4.6.3.1 and 4.4.6.3.2 are required.

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY.. At all times.

ACTION.. MODES 1, 2, 3, and 4

a. With any one or more chemistry parameters in excess of its Steady State Limit but within its Transient Limit, restore the Parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At all other times

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to ≤ 500 psig, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-10.

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	< 0.10 ppm*	≤ 1.00 ppm*
CHLORIDE	< 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.15 ppm	≤ 1.50 ppm

---

*Limit not applicable with $T_{avg} \leq 250°F$.

TABLE 4.4-10

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

CONTAMINANT	MINIMUM SAMPLING FREQUENCIES	MAXIMUM TIME BETWEEN SAMPLES
DISSOLVED OXYGEN	3 times per 7 days*	72 hours
CHLORIDE	3 times per 7 days	72 hours
FLUORIDE	3 times per 7 days	72 hours

---

*Not required with $T_{avg} \leq 250°F$.

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3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. ≤ 1.0 µCi/gram DOSE EQUIVALENT I-131, and

b. $\le 100/\bar{E}$ µCi/gram

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with $T_{avg} < 500°F$ within 6 hours.

b. With the specific activity of the primary coolant > $100/\bar{E}$ µCi/gram, be in HOT STANDBY with $T_{avg} < 500°F$ within 6 hours.

MODES 1, 2, 3, 4, and 5

a. With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ µCi/gram, perform the sampling analysis requirement of item 4a of Table 4.4-12 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the performance limits of the sampling and analysis program of Table 4.4-12.

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*With $T_{avg} \ge 500°F$.

TABLE 4.4-12

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SURVEILLANCE REQUIRED
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1,
3. Radiochemical for Ē Determination	1 per 6 months	1,
4. Isotopic Analysis for Iodine including I-131 I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100 /Ē μCi/gram, and	1#,2#,3#,4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored to within its limits.

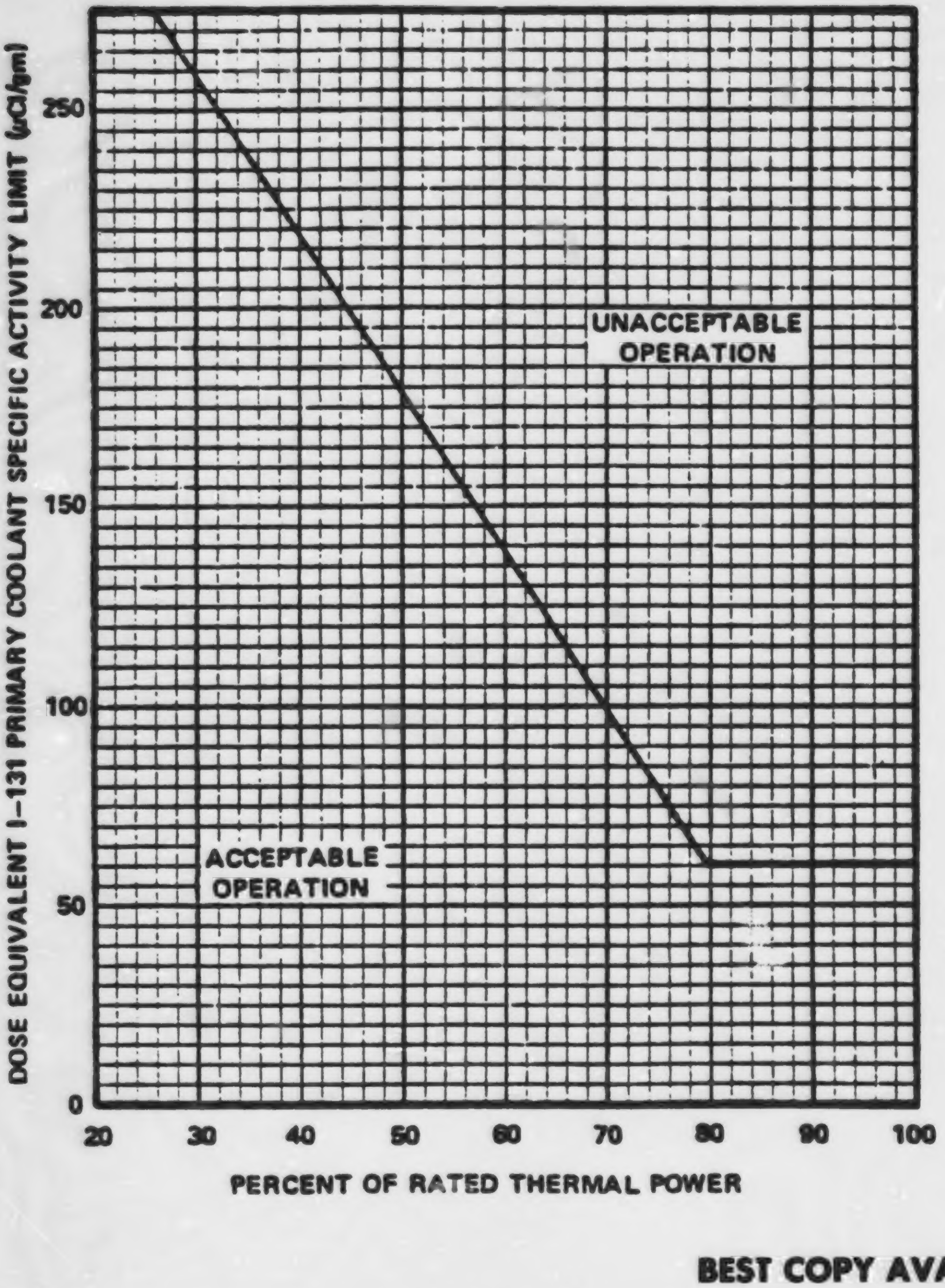

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FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 µCi/gram Dose Equivalent I-131

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of < 5°F in any 1-hour period during hydrostatic testing operations above system design pressure.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

MATERIAL PROPERTY BASIS

Controlling Material : Plate Metal
Copper Content : Conservatively Assumed to be 0.10 wt%
Phosphorus Content : 0.010 wt%
RT_{NDT} Initial : 60°F
RT_{NDT} After 10 EFPY : 1/4T, 139°F
3/4T, 114°F

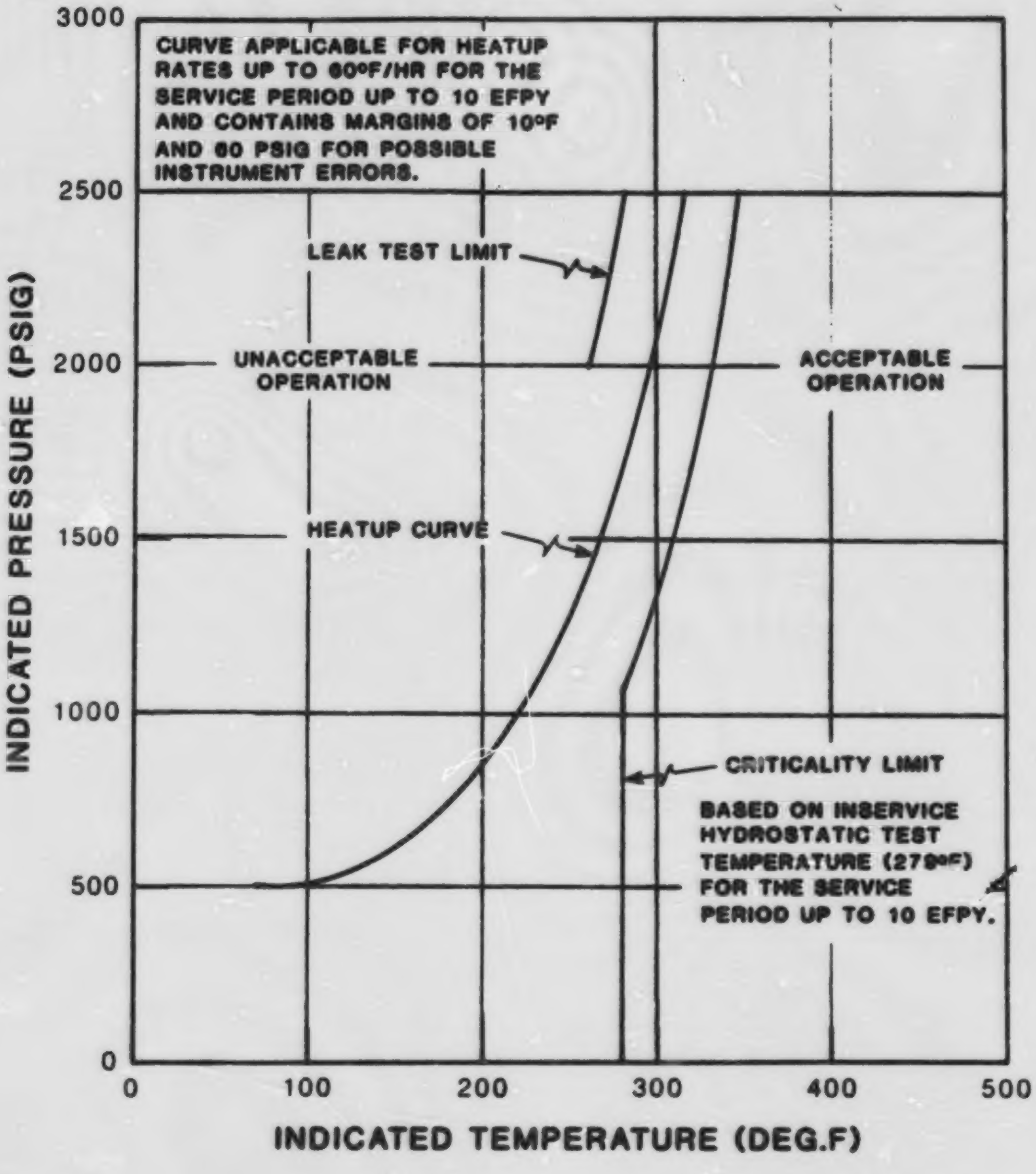
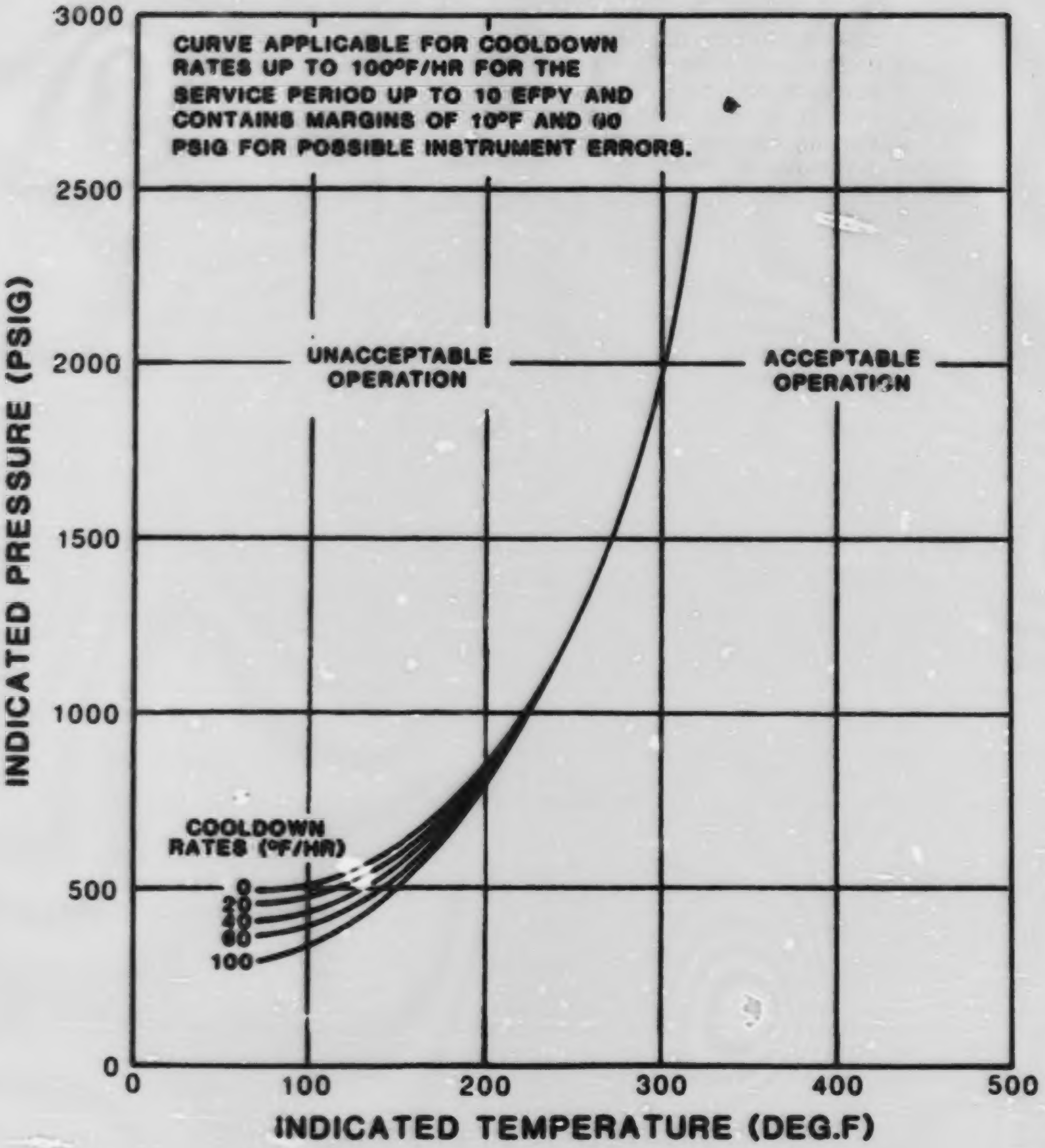


FIGURE 3.4-2 BEAVER VALLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 10 EFPY

MATERIAL PROPERTY BASIS

Controlling Material : Plate Metal
Copper Content : Conservatively Assumed to be 0.10 wt%
Phosphorus Content : 0.010 wt%
RT_{NDT} Initial : 60°F
RT_{NDT} After 10 EFPY : 1/4T, 139°F
3/4T, 114°F



CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

FIGURE 3.4-3 BEAVER VALLEY UNIT NO. 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 10 EFPY

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TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Capsule	Vessel Location	Lead Factor	Withdrawal Time (EFPY)	Estimated Capsule Fluence (n/cm^2)
U	343°	3.5	1st Refueling	0.8 x 10^{19}
V	107°	3.5	3	2.13 x 10^{19}
X	287°	3.5	6	4.26 x 10^{19}(a)
W	110°	2.9	11	6.48 x 10^{19}(b)
Y	290°	2.9	20	11.77 x 10^{19}
Z	340°	2.9	Standby	- -

(a) Approximate fluence at 1/4 T vessel wall thickness at end-of-life.

(b) Approximate fluence at vessel inner wall at end-of-life.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 200°F in any 1-hour period, and
- c. A maximum auxiliary spray water temperature differential of 625°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems (OPPS) shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with nominal maximum allowable lift settings which vary with the RCS temperature and which do not exceed the limits established in FIGURE 3.4-4, or

b. A Reactor Coolant System vent of $\geq$ 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the non-isolated RCS cold legs is $\leq$ 350°F.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status. Refer to Technical Specification 3.4.1.6 for further limitations.

b. With both PORVs inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE BY:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

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SURVEILLANCE REQUIREMENTS(Continued)

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. Stroking the operable PORV(s) each time the plant enters MODE 5, unless tested within the preceding 3 months.

4.4.9.3.2 The > 3.14 square inch RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

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*Except when the vent pathway is provided with a valve which is locked, or provided with remote position indication, sealed, or otherwise secured in the open position, then verify these valves open at least once per 7 days.

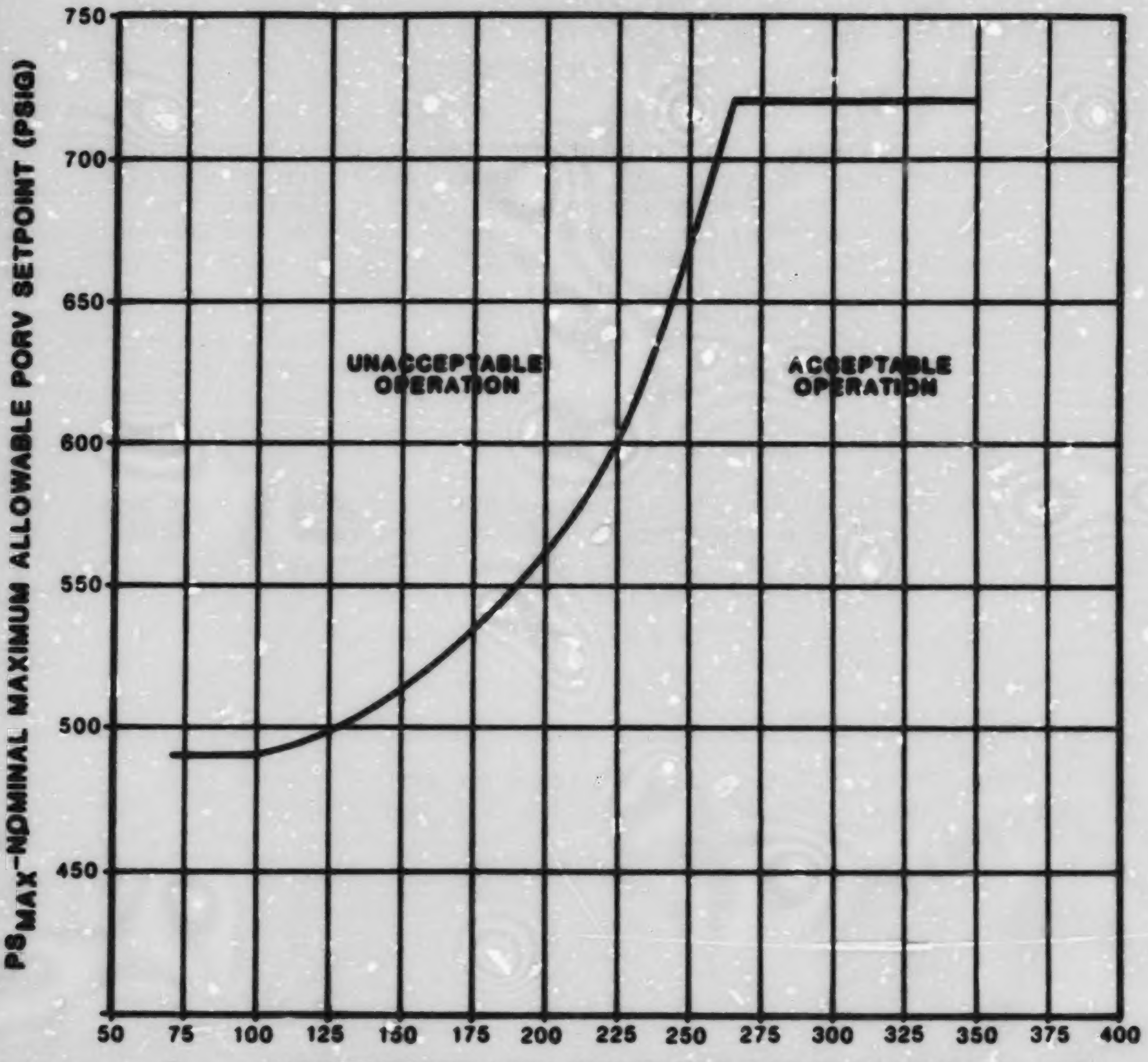

Figure 3.4-4
Maximum Allowable Nominal PORV Setpoint
For The Overpressure Protection System

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REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2, AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 Each ASME Code Class 1, 2, and 3 component shall be demonstrated OPERABLE in accordance with Specification 4.0.5.

3/4.4.11 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.11 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or two PORV(s) inoperable as a result of causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s).

c. With all three PORVs inoperable due to causes other than excessive seat leakage, within 72 hours either restore one PORV to OPERABLE status or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV; and (2) apply the ACTION b. or c. above, as appropriate, for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and
b. Operating the valve through one complete cycle of full travel.

4.4.11.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b., c. or d. in Specification 3.4.11.

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REACTOR COOLANT SYSTEM HEAD VENTS

LIMITING CONDITION FOR OPERATION

3.4.12 All Reactor Coolant System head vent valves, powered from emergency buses shall be OPERABLE* and closed** for each of the reactor vessel head vent paths.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With at least one vent path from the above location OPERABLE and one or more power operated vent valves inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable valve(s) is maintained closed with power removed. Power operation may continue until the next scheduled outage, at which time all Reactor Coolant System head vent valves shall be OPERABLE prior to entry into MODE 1. The provisions of Specification 3.0.4 are not applicable.

b. With all vent paths from the above location inoperable maintain the inoperable valves closed with power removed or close the manual isolation valves, and restore at least one vent path from the above locations to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 Each Reactor Coolant System head vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the manual isolation valve in the vent path is locked or sealed in the open position.
2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room.
3. Verifying flow through the Reactor Coolant System Head vent path to the Pressurizer Relief Tank.

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*For purposes of this Specification, an inoperable vent valve is defined as: a valve which exhibits leakage in excess of Specification 3.4.6.2 limits, or cannot be opened and closed on demand, or does not have its normal emergency power supply OPERABLE.

**These valves may be operated for required venting operations and leak testing in MODES 3 and 4.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Between 7532 and 7802 gallons of borated water,

c. Between 1900 and 2100 ppm of boron, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

   1. Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

   2. Verifying that each accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution.

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*Pressurizer Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator control circuit is disconnected by removal of the plug in the lock out jack from the circuit.

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

   1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

   2) Upon receipt of a Safety Injection test signal.

4.5.1.2 Each accumulator water level and pressure alarm channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of a CHANNEL FUNCTIONAL TEST.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

4.5.1.3 During normal plant cooldown and depressurization, each accumulator discharge isolation valve 2 SIS-MOV 865 A, B and C shall be verified to be closed and de-energized when RCS pressure is reduced to 1,000 ± 100 psig.

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two separate and independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump, and

c. One OPERABLE recirculation spray pump* capable of supplying the safety injection flow path during recirculation phase, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.**

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operator control circuits disconnected by removal of the plug in the lock out circuit from each circuit:

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*Recirculation spray pump 2RSS-P21C or 2RSS-P21D.

**The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pumps declared inoperable pursuant to Specification 4.5.3.2 provided the centrifugal charging pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

	Valve Number	Valve Function	Valve Position
a.	2SIS-MOV 8889	LHSI to hot legs	Closed
b.	2SIS-MOV 869A	HHSI to hot leg	Closed
c.	2SIS-MOV 869B	HHSI to hot leg	Closed
d.	2SIS-MOV 841	HHSI to cold leg	Open
e.	2CHS-MOV 8132A	HHSI pump disch x-conn	Open
f.	2CHS-MOV 8132B	HHSI pump disch x-conn	Open
g.	2CHS-MOV 8133A	HHSI pump disch x-conn	Open
h.	2CHS-MOV 8133B	HHSI pump disch x-conn	Open

b. By verifying that each of the following pumps develop the required differential pressure on recirculation flow when tested pursuant to Specification 4.0.5.

1.)	Centrifugal charging pump	$\geq$ 2437 psid
2.)	Low head safety injection pump,	$\geq$ 103 psid

c. At least once per 31 days by:

1) Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

2) Verifying that each ECCS subsystem is aligned to receive electrical power from separate OPERABLE emergency buses.

SURVEILLANCE REQUIREMENTS (Continued)

d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

    1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

    2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

e. At least once per 18 months by:

    1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

f. At least once per 18 months, during shutdown, by:

    1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

    2. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection signal.

    3. Verifying that the centrifugal charging pump and low head safety injection pumps start automatically upon receipt of a safety injection signal.

g. The containment recirculation spray subsystem shall be demonstrated OPERABLE per the applicable portions of Specification 4.6.2.2.

ECCS SUBSYSTEMS - $T_{avg} < 350°F$

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Low Head Safety Injection Pump, and

c. One OPERABLE recirculation spray pump* capable of supplying the safety injection flow path during recirculation phase, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days describing the circumstances of the actuation and the total accumulated actuation cycle to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable** by verifying that the control switches are placed in the PULL-TO-LOCK position and tagged within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first, and at least once per 12 hours thereafter.

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*Recirculation spray pump 2RSS-P21C or 2RSS-P21D.

**An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

  1. All penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

  2. All equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.10 percent by weight of the containment air per 24 hours at P_a, (44.7 psig).

b. A combined leakage rate of $< 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests as identified in Table 3.6-1, when pressurized to P_a (44.7 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972:

a. A Type-A test (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10-month intervals during shutdown at P_a (44.7 psig).

b. If any Periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a.

   2. Has a duration sufficient to accurately establish the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a (44.7 psig).

d. Type B and C tests shall be conducted with gas at P_a^* (44.7 psig) at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Penetrations using continuous leakage monitoring systems, and

   3. Valves pressurized with fluid from a seal system.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the Provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (49.2 psig) and the seal system capacity is adequate to maintain system pressure for at least 30 days.

g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to determine the inaccuracy of the measured leakage rates due to maximum measurement accuracy and instrument repeatability; the measured leakage rates shall be adjusted to include the measurement error.

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*Applicable valves may be tested using water as the pressure fluid in accordance with the Inservice Testing Program

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a (44.7 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain the associated OPERABLE air lock door closed and either restore the associated inoperable air lock door to OPERABLE status within 24 hours or lock the associated OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the associated OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With a containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each containment entry, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage when the gap between the door seals is pressurized for at least 2 minutes to:

1. Personnel airlock ≥ 44.7 psig

2. Emergency air lock ≥ 10.0 psig

or, by quantifying the total air lock leakage to insure the requirements of 3.6.1.3.b are met.

b. By conducting overall air lock leakage tests, at not less than P_a (44.7 psig), and verifying the overall air lock leakage rate is within its limit:

1. At least once per 6 months, # and

2. Upon completion of maintenance which has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 18 months during shutdown verifying:

1. Only one door in each air lock can be opened at a time, and

2. No detectable seal leakage when the volume between the emergency air lock shaft seals is pressurized to greater than or equal to 44.7 psig for at least 2 minutes.

# The provisions of Specification 4.0.2 are not applicable.

* Exemption of Appendix J of 10 CFR 50

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary Containment internal air partial pressure shall be maintained > 9.0 psia and within the acceptable operation range (below and to the left of the RWST water temperature limit line) shown on Figure 3.6-1 as a function of service water temperature.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal air partial pressure < 9.0 psia or above the RWST water temperature limit line shown on Figure 3.6-1, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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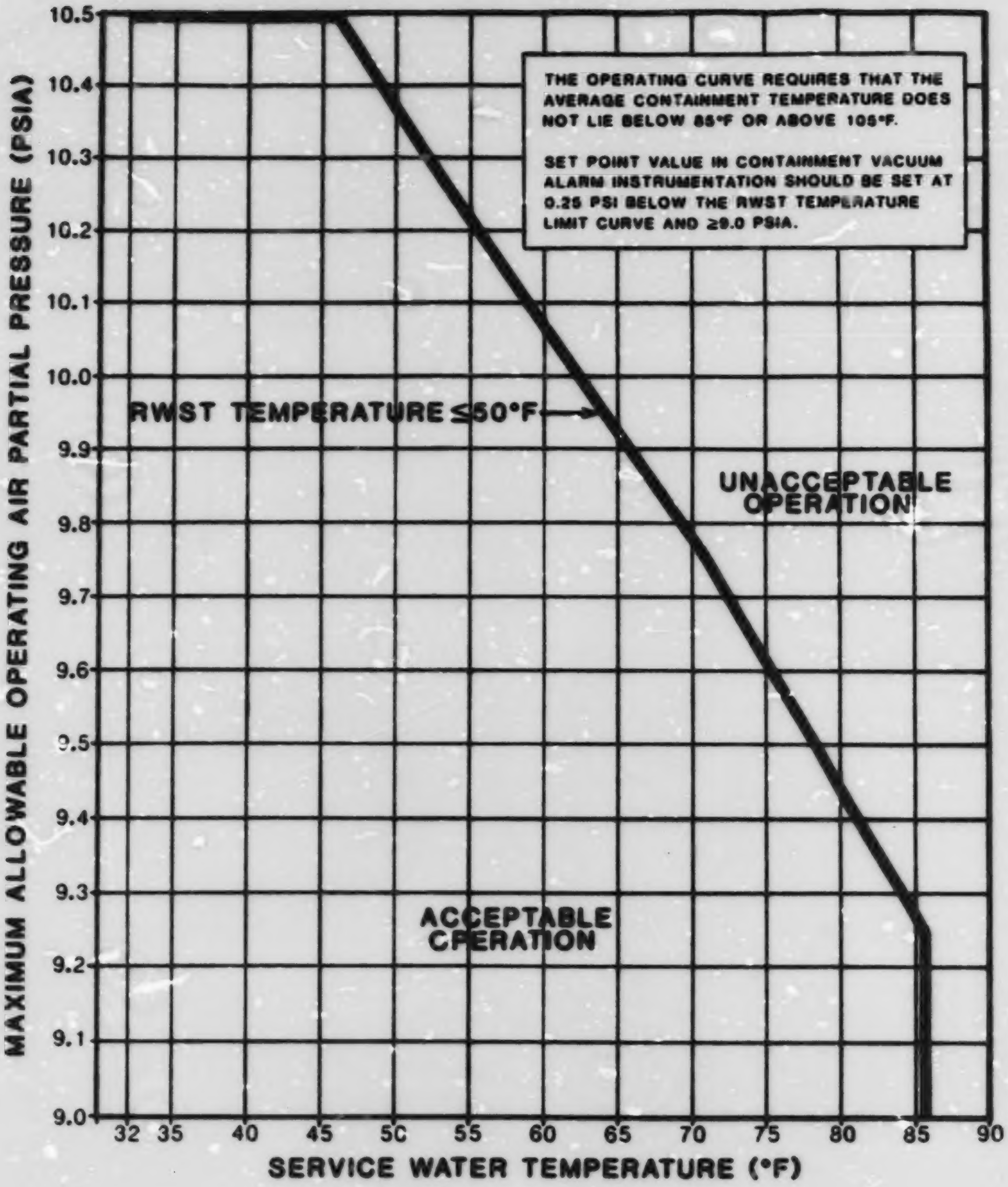

THE OPERATING CURVE REQUIRES THAT THE AVERAGE CONTAINMENT TEMPERATURE DOES NOT LIE BELOW 85°F OR ABOVE 105°F.

SET POINT VALUE IN CONTAINMENT VACUUM ALARM INSTRUMENTATION SHOULD BE SET AT 0.25 PSI BELOW THE RWST TEMPERATURE LIMIT CURVE AND ≥9.0 PSIA.

FIGURE 3.6-1

MAXIMUM ALLOWABLE PRIMARY CONTAINMENT AIR PRESSURE VERSUS SERVICE WATER TEMPERATURE AND RWST WATER TEMPERATURE

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained $\leq$ 105°F and $\geq$ 85°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 105°F or < 85°F restore the average air temperature to within the limit within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average maximum and minimum air temperatures shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours. The nearest alternate detector may be used for temperature determination up to a maximum of one per location.

Location

a. Reactor Head Storage Area - Elev. 802'-0"

b. Pressurizer Cubicle - Elev. 802'0"

c. RC Annulus - Elev. 777'-4"

d. RHR Heat Exchanger - Elev. 801'-6"

e. RC Annulus - Elev. 701'-6"

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CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Liner Plate and Concrete The structural integrity of the containment liner plate and concrete shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by:

a. a visual inspection of the accessible surfaces and verifying no apparent changes in appearance or other abnormal degradation.

b. a visual inspection of accessible containment liner test channels prior to each Type A containment leakage rate test. Any containment liner test channel which is found to be damaged to the extent that channel integrity is impaired or which is discovered with a vent plug removed, shall be removed and a protective coating shall be applied to the liner in that area.

c. a visual inspection of the dome area prior to each Type A containment leakage rate test to insure the integrity of the protective coating.

4.6.1.6.2 Reports An initial report of any abnormal degradation of the containment structure detected during the above required tests and inspections shall be made within 10 days after completion of the surveillance requirements of this specification, and the detailed report shall be submitted pursuant to Specification 6.9.2 within 90 days after completion. This report shall include a description of the condition of the liner plate and concrete, the inspection procedure, the tolerances on cracking and the corrective actions taken.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two separate and independent containment quench spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one containment quench spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

2. Verifying the temperature of the borated water in the refueling water storage tank is within the limits of Specification 3.1.2.8.b.3.

b. By verifying, that on a recirculation flow, each pump develops a differential pressure of $\geq$ 138 psid at a flow of $\geq$ 3000 gpm when tested pursuant to Specification 4.0.5.

c. At least once per 18 months during shutdown, by:

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a test signal.

3. Verifying that each spray pump starts automatically on a test signal.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Four separate and independent containment recirculation spray subsystems, each composed of a spray pump, associated heat exchanger and flow path shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. For subsystems containing recirculation spray pumps 2RSS-P21A or 2RSS-P21B: With one containment recirculation spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

b. For subsystems containing recirculation spray pumps 2RSS-P21C or 2RSS-P21D: See action statements in Specification 3.5.2 or 3.5.3.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each containment recirculation spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position;

b. When tested pursuant to Specification 4.0.5, manually start each recirculation spray pump and verify the pump shaft rotates;

c. At least once per 18 months by verifying that on a Containment Pressure-High-High signal, each recirculation spray pump starts automatically after a 628 ± 5 second delay.

d. At least once per 18 months, during shutdown, by verifying, that on recirculation flow, each recirculation spray pump develops a differential pressure of $\geq$ 112 psid at a flow of $\geq$ 3500 gpm.

e. At least once per 18 months during shutdown, by:

   1. Cycling each power operated (excluding automatic) valve in the flow path not testable during plant operation, through at least one complete cycle of full travel.

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2. Verifying that each automatic valve in the flow path actuates to its correct position on a test signal.

3. Initiating flow through each Service Water subsystem and its two associated recirculation spray heat exchangers, and verifying a flow rate of at least 12,000 gpm.

f. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT SYSTEMS

CHEMICAL ADDITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The chemical addition system shall be OPERABLE with:

a. A chemical addition tank containing at least 8500 gallons of between 23 and 25 percent by weight NaOH solution, and

b. Two chemical injection pumps each capable of adding NaOH solution from the chemical addition tank to a containment quench spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the chemical addition system inoperable, restore the system to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore the chemical addition system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The chemical addition system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying that on recirculation flow, each injection pump develops a flow between 40 and 60 gpm when tested pursuant to Specification 4.0.5.

c. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and
2. Verifying the concentration of the NaOH solution by chemical analysis.

d. At least once per 18 months, during shutdown, by:

1. Cycling each valve in the chemical addition system flow path that is not testable during plant operation, through at least one complete cycle of full travel.
2. Verifying that each automatic valve in the flow path actuates to its correct position on a test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6.1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate the affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 6 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE:

a. At least once per 92 days by:

1. Cycling each OPERABLE power operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

2. Cycling each weight or spring loaded check valve testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is $<$ 1.2 psid and opens when the differential pressure in the direction of flow is $\geq$ 1.2 psid but less than 6.0 psid.

b. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the applicable cycling test, above, and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position.

d. Cycling each power operated or automatic valve through at least one complete cycle of full travel and measuring the isolation time pursuant to Specification 4.0.5.

e. Cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is $<$ 1.2 psid and opens when the differential pressure in the direction of flow is $\geq$ 1.2 psid but less than 6.0 psid.

f. Cycling each manual valve not locked, sealed or otherwise secured in the closed position through at least one complete cycle of full travel.

TABLE 3.6-1

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
1	Comp Cool from Res Heat Exch	(1)(B) 2CCP-MOV157-2 2CCP-RV105	< 60 N/A	(1)(B) 2CCP-MOV157-1	< 60
2	Comp Cool to Res Heat Exch	(1)(B) 2CCP-MOV150-2 2CCP-RV102	< 60 N/A	(1)(B) 2CCP-MOV150-1	< 60
4	Comp Cool to Res Heat Exch	(1)(B) 2CCP-MOV151-2 2CCP-RV103	< 60 N/A	(1)(B) 2CCP-MOV151-1	< 60
5	Comp Cool from Res Heat Exch	(1)(B) 2CCP-MOV156-2 2CCP-RV104	< 60 N/A	(1)(B) 2CCP-MOV156-1	< 60
6	SPARE				
7	High Head Safety Injection	(3)(2) 2SIS-83	N/A	(3)(2) 2SIS-MOV869A	N/A
9	SPARE				
11	Instrument Air	(A) 2IAC-MOV133	< 60	(A) 2IAC-MOV134	< 60
13	SPARE				
14	Chill & Service Wtr to Cont. Air Recirc Cooling coils	(B) 2SWS-MOV153-2	< 60	(B) 2SWS-MOV153-1 2SWS-RV153	< 60 N/A
15	CHARGING	(3)(2) 2CHS-31	N/A	(3)(2) 2CHS-MOV289	< 10
16	SPARE				

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE		MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE		MAXIMUM STROKE TIME (SEC)
17	High Head Safety Injection	(3)(2)	2SIS-84	N/A	(3)(2)	2SIS-MOV869B	N/A
19	Seal Water from Reactor Coolant Pump	(A)	2CHS-MOV378	< 60	(A)	2CHS-MOV381	< 60
			2CHS-473	N/A			
20	Safety Injection Accumulator Makeup		2SIS-42	N/A	(1)	2SIS-41	N/A
						2SIS-RV130	N/A
21	Chill & Service Wtr from Cont. Air Recirc Cooling Coils	(B)	2SWS-MOV155-2	< 60	(B)	2SWS-MOV155-1	< 60
						2SWS-RV155	N/A
22	SPARE						
23	SPARE						
24	Residual Heat Removal to Refueling Water Tank		2RHS-107	N/A		2RHS-15	N/A
						2RHS-RV100	N/A
25	Chill & Service Wtr from Cont. Air Recirc Cooling Coils	(B)	2SWS-MOV154-2	< 60	(B)	2SWS-MOV154-1	< 60
						2SWS-RV154	N/A
27	Chill & Service Wtr to Cont. Air Recirc Cooling Coils	(B)	2SWS-MOV152-2	< 60	(B)	2SWS-MOV152-1	< 60
						2SWS-RV152	N/A
28	Reactor Coolant Letdown	(A)	2CHS-AOV200A	10	(A)	2CHS-AOV204	< 60
		(A)	2CHS-AOV200B	10			
		(A)	2CHS-AOV200C	10			
		(1)	2CHS-HCV142	N/A			
			2CHS-RV203	N/A			

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TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE		MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE		MAXIMUM STROKE TIME (SEC)
29	Pri Dr. Trans Pump Disch	(A)	2DGS-AOV108A	< 60	(A)	2DGS-AOV108B 2DGS-RV115	< 60 N/A
30	SPARE						
31	SPARE						
32	SPARE						
33	SPARE						
34	High Head Injection Line	(3)(2)	2SIS-94	N/A	(3)(2) (3)(2)	2SIS-MOV836 2SIS-MOV840	N/A N/A
35	Inj Seal Wtr to Reactor Coolant Pump	(3)(2)	2CHS-474	N/A	(2)(3)	2CHS-MOV308A	N/A
36	Inj Seal Wtr to Reactor Coolant Pump	(3)(2)	2CHS-476	N/A	(2)(3)	2CHS-MOV308B	N/A
37	Inj Seal Wtr to Reactor Coolant Pump	(3)(2)	2CHS-475	N/A	(2)(3)	2CHS-MOV308C	N/A
38	Sump Pump Discharge	(A)	2DAS-AOV100A	< 60	(A)	2DAS-AOV100B 2DAS-RV110	< 60 N/A
39	St Gen Blowdown		Closed System	N/A	(2)	2BDG-AOV100A-1	< 60
40	St Gen Blowdown		Closed System	N/A	(2)	2BDG-AOV100B-1	< 60
41	St Gen Blowdown		Closed System	N/A	(2)	2BDG-AOV100C-1	< 60

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE		MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE		MAXIMUM STROKE TIME (SEC)
42	Service Air		2SAS-15	N/A		2SAS-14	N/A
43	Air Monitor Sample		2CVS-93	N/A	(A)	2CVS-SOV102	< 60
44	Air Monitor Sample	(1)(A)	2CVS-SOV153B	< 60	(1)(A)	2CVS-SOV153A	< 60
45	Primary Grade Water		2RCS-72	N/A	(A)	2RCS-AOV519 2RCS-RV100	< 60 N/A
46	Loop Fill	(3)(2)	2CHS-472	N/A	(3)(2)(1)	2CHS-FCV 160	N/A
47	SPARE						
48	Primary Vent Header	(A)	2VRS-AOV109A-2	< 60	(A)	2VRS-AOV109A-1	< 60
49	Nitrogen Supply Manifold		2RCS-68	N/A	(A)	2RCS-AOV101	< 60
50	SPARE						
51	SPARE						
52	SPARE						
53	Nitrogen Manifold	(A)	2GNS-AOV101-2	< 10	(A)	2GNS-AOV101-1	< 60

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE		MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE		MAXIMUM STROKE TIME (SEC)
55	Leakage Detection				(2)	2LMS-SOV953	< 60(4)
	Press Relief Tank	(1)(A)	2SSR-SOV130A-1	< 60	(1)(A)	2SSR-SOV130A-2	< 60
	Accumulator Water Sample	(A)	2SSR-AOV109A-1	< 60	(A)	2SSR-AOV109A-2 2SSR-RV117	< 60 N/A
	Hydrogen Analyzer	(1)	2HCS-SOV136A	N/A	(1)	2HCS-SOV136B	N/A
56	Cold Leg Sample	(A)	2SSR-AOV102A-1	< 60	(A)	2SSR-AOV102A-2 2SSR-RV118	< 60 N/A
	Hot Leg Sample	(1)(A)	2SSR-SOV128A-1	< 60	(1)(A)	2SSR-SOV128A-2 2SSR-RV120	< 60 N/A
	Pressurizer Liquid Space Sample	(A)	2SSR-AOV100A-1	< 60	(A)	2SSR-AOV100A-2 2SSR-RV119	< 60 N/A
	Blowdown Sample		Closed System	N/A	(2)	2SSR-AOV117A	< 60
57	Leak Detection				(2)	2LMS-SOV950	< 60(4)
	Blowdown Sample		Closed System	N/A	(2)	2SSR-AOV117B	< 60
	Pressurizer Vapor Space Sample	(A)	2SSR-AOV112A-1	< 60	(A)	2SSR-AOV112A-2 2SSR-RV121	< 60 N/A
	Hydrogen Analyzer	(1)	2HCS-SOV135A	N/A	(1)	2HCS-SOV135B	N/A
59	Instrument Air Containment		2IAC-22	N/A	(A)	2IAC-MOV130	< 60

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE		MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE		MAXIMUM STROKE TIME (SEC)
60	Low Head Safety Injection Discharge	(3)(2)	2SIS-132	N/A	(3)(2)	2SIS-MOV8888B	N/A
61	Low Head Safety Injection Discharge	(3)(2)	2SIS-130	N/A	(3)(2)	2SIS-MOV8889	N/A
62	Low Head Safety Injection Discharge	(3)(2)	2SIS-133	N/A	(3)(2)	2SIS-MOV8888A	N/A
63	Quench Pump Discharge		2QSS-4	N/A	(B)	2QSS-MOV101A 2QSS-RV101A	< 60 (4) N/A
64	Quench Pump Discharge		2QSS-3	N/A	(B)	2QSS-MOV101B 2QSS-RV101B	< 60 (4) N/A
65	Fuel Transfer Tube	(7)	Flange	N/A			
66	Recirc Spray Pump Suction				(B)(2)	2RSS-MOV155A	< 60 (4)
67	Recirc Spray Pump Suction				(B)(2)	2RSS-MOV155C	< 60 (4)
68	Recirc Spray Pump Suction				(B)(2)	2RSS-MOV155D	< 60 (4)
69	Recirc Spray Pump Suction				(B)(2)	2RSS-MOV155B	< 60 (4)
70	Recirculation Pump Discharge	(2)	2RSS-29	N/A	(B)(2) (6)	2RSS-MOV156A 2RSS-RV156A	< 60 (4) N/A
71	Recirculation Pump Discharge	(2)	2RSS-31	N/A	(10)(B)(2) (6)	2RSS-MOV156C 2RSS-RV156C	< 60 (4) N/A

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
73	Main Steam System "A"	Closed System	N/A	(2) 2MSS-HYV101A	5
		Closed System	N/A	(2) 2MSS-AOV102A	N/A
		Closed System	N/A	(2) 2MSS-SOV105A	N/A
		Closed System	N/A	(6) 2MSS-SV101A	N/A
		Closed System	N/A	(6) 2MSS-SV102A	N/A
		Closed System	N/A	(6) 2MSS-SV103A	N/A
		Closed System	N/A	(6) 2MSS-SV104A	N/A
		Closed System	N/A	(6) 2MSS-SV105A	N/A
	Steam Drains System	Closed System	N/A	(2) 2SDS-AOV111A-1	< 60
		Closed System	N/A	(2) 2SDS-AOV129B	< 60
	Steam Vent System	Closed System	N/A	(6) 2SVS-PCV101A	N/A
		Closed System	N/A	(6) 2SVS-HCV104	N/A

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
74	Main Steam System "B"	Closed System	N/A	(2) 2MSS-HYV101B	5
		Closed System	N/A	(2) 2MSS-AOV102B	N/A
		Closed System	N/A	(2) 2MSS-SOV105B	N/A
		Closed System	N/A	(6) 2MSS-SV101B	N/A
		Closed System	N/A	(6) 2MSS-SV102B	N/A
		Closed System	N/A	(6) 2MSS-SV103B	N/A
		Closed System	N/A	(6) 2MSS-SV104B	N/A
		Closed System	N/A	(6) 2MSS-SV105B	N/A
	Steam Drains System	Closed System	N/A	(2) 2SDS-AOV111B-1	< 60
		Closed System	N/A	(2) 2SDS-AOV129B	< 60
	Steam Vent System	Closed System	N/A	(6) 2SVS-PCV101B	N/A
		Closed System	N/A	(6) 2SVS-HCV104	N/A

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
75	Main Steam System "C"	Closed System	N/A	(2) 2MSS-HYV101C	5
		Closed System	N/A	(2) 2MSS-AOV102C	N/A
		Closed System	N/A	(2) 2MSS-SOV105C	N/A
		Closed System	N/A	(6) 2MSS-SV101C	N/A
		Closed System	N/A	(6) 2MSS-SV102C	N/A
		Closed System	N/A	(6) 2MSS-SV103C	N/A
		Closed System	N/A	(6) 2MSS-SV104C	N/A
		Closed System	N/A	(6) 2MSS-SV105C	N/A
	Steam Drains System	Closed System	N/A	(2) 2SDS-AOV111C-1	< 60
		Closed System	N/A	(2) 2SDS-AOV129B	< 60
	Steam Vent System	Closed System	N/A	(6) 2SVS-PCV101C	N/A
		Closed System	N/A	(6) 2SVS-HCV104	N/A
76	Feedwater "A"	Closed System	N/A	(2) 2FWS-HYV157A (2) 2FWS-28	5 N/A
77	Feedwater "B"	Closed System	N/A	(2) 2FWS-HYV157B (2) 2FWS-29	5 N/A
78	Feedwater "C"	Closed System	N/A	(2) 2FWS-HYV157C (2) 2FWS-30	5 N/A

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
79	Aux Feed "A"	(2) 2FWE-99	N/A	(2) 2FWE-HCV100E	N/A
				(2) 2FWE-HCV100F	N/A
				(2) 2FWE-42A	N/A
				(2) 2FWE-42B	N/A
80	Aux Feed "B"	(2) 2FWE-100	N/A	(2) 2FWE-HCV100C	N/A
				(2) 2FWE-HCV100D	N/A
				(2) 2FWE-43A	N/A
				(2) 2FWE-43B	N/A
83	Aux Feed "C"	(2) 2FWE-101	N/A	(2) 2FWE-HCV100A	N/A
				(2) 2FWE-HCV100B	N/A
				(2) 2FWE-44A	N/A
				(2) 2FWE-44B	N/A
87	Hydrogen Recombiner Discharge		N/A	(1) 2HCS-MOV117	N/A
				(1) 2HCS-111	N/A
88	Hydrogen Recombiner Discharge		N/A	(1) 2HCS-MOV116	N/A
				(1) 2HCS-110	N/A
89	SPARE				
90	Purge Duct Exhaust	(5) 2HVR-MOD23B	10	(5) 2HVR-MOD23A	10
91	Purge Duct Supply	(5) 2HVR-MOD25B	10	(5) 2HVR-MOD25A	10
				(5) 2HVR-DMP206	N/A
92	Hydrogen Recombiner Isolation			(1) 2HCS-SOV114B	N/A
				(1) 2HCS-SOV115B	N/A
	Reactor Cont. Vacuum Pump Suction			(A) 2CVS-SOV151B	< 60
				(A) 2CVS-SOV152B	< 60

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TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE		MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE		MAXIMUM STROKE TIME (SEC)
93	Hydrogen Recombiner Isolation				(1) (1)	2HCS-SOV114A 2HCS-SOV115A	N/A N/A
	Reactor Cont. Vacuum Isolation				(A) (A)	2CVS-SOV151A 2CVS-SOV152A	< 60 < 60
94	Ejector Suction		2CVS-151	N/A		2CVS-151-1	N/A
96	SPARE						
97	Leakage Detection				(2)	2LMS-SOV952	< 60 (4)
	Blowdown Sample		Closed System	N/A	(2)	2SSR-AOV117C	< 60
	Liquid Sample - Cont. Sump & RHS	(1)(A)	2SSR-SOV129A-1	< 60	(1)(A)	2SSR-SOV129A-2 2SSR-RV122	< 60 N/A
	Hydrogen Analyzer	(1)	2HCS-SOV133B	N/A	(1)	2HCS-SOV134B	N/A
98	SPARE						
99	Hose Rack Supply		2FPW-761	N/A	(A)	2FPW-AOV206	< 60
100	SPARE						
101	Reactor Cont. Deluge - Cable Pent. Area & RHS Pump		2FPW-753	N/A	(A)	2FPW-AOV205	< 60
103	Reactor Cavity Purif Inlet		2FNC-121	N/A		2FNC-38	N/A
104	Reactor Cavity Purif Outlet		2FNC-122	N/A		2FNC-9	N/A

TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE		MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE		MAXIMUM STROKE TIME (SEC)
105	Leak Detection				(2)	2LMS-SOV951	< 60 (4)
	Leak Detection					2LMS-51 2LMS-52	N/A N/A
	Hydrogen Analyzer	(1)	2HCS-SOV133A	N/A	(1)	2HCS-SOV134A	N/A
	Post Accident Sampling	(A)(1)	2PAS-SOV105A-1	< 60	(A)(1)	2PAS-SOV105A-2	< 60
106	Safety Inj. Test Line	(A)	2SIS-MOV842	< 60	(A)	2SIS-AOV889 2SIS-RV175	< 60 N/A
108	SPARE						
110	SPARE						
113	Safety Injection	(3)(2)	2SIS-95	N/A	(3)(2) (3)(2)	2SIS-MOV867C 2SIS-MOV867D	< 10 (4) < 10 (4)
114	Recirculation Pump Discharge	(2)	2RSS-32	N/A	(10)(B)(2) (6)	2RSS-MOV156D 2RSS-RV156D	< 60 (4) N/A
115	Recirculation Pump Discharge	(2)	2RSS-30	N/A	(B)(2) (6)	2RSS-MOV156B 2RSS-RV156B	< 60 (4) N/A
116	Fire Protection HVR Filter B		2FPW-388	N/A	(A)	2FPW-AOV221	< 60
117	Fire Protection HVR Filter A		2FPW-382	N/A	(A)	2FPW-AOV204	< 60

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TABLE 3.6-1 (Cont)

CONTAINMENT PENETRATIONS

PENT. NO.-AREA	IDENTIFICATION/DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
118	Quench Spray System	2QSS-267	N/A	(11)(B)(1) 2QSS-SOV100A	N/A
				(11)(B)(1) 2QSS-SOV100B	N/A
119	RVLIS	Note 12	N/A	Note 12	N/A
Primary Containment Personnel Air Lock 2 PHS-PAL 1					
	Equalizing Valve	(7) 2PHS-112	N/A		
	Equalizing Valve	(7) 2PHS-113	N/A		
	Equalizing Valve	(7) 2PHS-101	N/A		
	Equalizing Valve			(7) 2PHS-110	N/A
	Equalizing Valve			(7) 2PHS-111	N/A
	Equalizing Valve			(7) 2PHS-100	N/A
Emergency Containment Air Lock 2PHS-EAL 1					
	Equalizing Valve	(7) 2PHS-202	N/A		
	Equalizing Valve			(7) 2PHS-201	N/A

TABLE 3.6-1 (Cont)

NOTES:

(A) Containment Isolation Phase A.
(B) Containment Isolation Phase B.

(1) May be opened on an intermittent basis under administrative control.

(2) Not subject to Type C leakage tests.

(3) May be leakage tested with water as the test fluid.

(4) Maximum opening time.

(5) Applicability: During CORE ALTERATIONS or movement of irradiated fuel within containment. The provisions of Specification 3.0.4 are not applicable. The containment Purge Exhaust and Supply valves will be locked shut during operation in modes 1, 2, 3, and 4.

(6) Not subject to the requirements of Specification 3/4.6.3. Listed in Table 3.6-1 for information only.

(7) Tested under Type "B" testing.

(8) Temporarily removed and penetration plugged.

(9) Auto open on Safety Injection recirculation signal.

(10) Auto close on Safety Injection recirculation signal.

(11) Auto open on QSS switchover signal.

(12) Isolation is provided by bellows operated hydraulic isolators.

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3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two separate and independent wide-range containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one wide-range hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

b. With both wide-range hydrogen analyzers inoperable, restore at least one wide-range hydrogen analyzer to OPERABLE status within 72 hours or be in HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by:

a. Performing a CHANNEL CALIBRATION using sample gases containing:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two separate and independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test using outside atmospheric flow rate of $>$ 42 scfm that the heater outlet temperature increases to $\geq$ 700°F within 90 minutes and is maintained for at least 2 hours.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners (i.e., loose wiring or structural connections, deposits of foreign materials, etc.).

   3. Verifying during a recombiner system functional test using containment atmospheric air at a pressure of $\leq$ 13 psia and a flow rate of $\geq$ 42 scfm, that the heater temperature increases to $\geq$ 1100°F within 5 hours and is maintained for at least 4 hours.

   4. Verifying the integrity of all heater electrical circuits by performing a continuity and resistance to ground test immediately following the above-required functional test. The resistance to ground for any heater phase shall be $\geq$ 10,000 ohms.

c. Verifying that the hydrogen recombiner isolation valves (2HCS-MOV110A&B and 2HCS-MOV113A&B) are closed and de-energized after every surveillance test (per 4.6.4.2.a) is completed or after their use, post-accidents, to recombine hydrogen in the containment is completed.

3/4.6.4.3 (This specification number is not used)

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

3.6.5.1 The inside and outside manual isolation valves in the steam jet air ejector suction line shall be closed.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the inside or outside manual isolation valve in the steam jet air ejector suction line not closed, restore the valve to the closed position within 1 hour or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 The steam jet air ejector suction line outside manual isolation valve shall be determined to be in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 350°F and at least once per 31 days thereafter.

4.6.5.1.2 The steam jet air ejector suction line inside manual isolation valve shall be determined to be sealed or locked in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 350°F.

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3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings and orifice sizes as shown in Table 3.7-2 by performance of the surveillances required by Specification 4.0.5.

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TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	65
3	43

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (± 1%)	ORIFICE DIAMETER
a. 2MSS-SV101A, B & C	1075 psig	4.515 in.
b. 2MSS-SV102A, B & C	1085 psig	4.515 in.
c. 2MSS-SV103A, B & C	1095 psig	4.515 in.
d. 2MSS-SV104A, B & C	1110 psig	4.515 in.
e. 2MSS-SV105A, B & C	1125 psig	4.515 in.

---

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two capable of being powered from separate emergency busses and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7 1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that:

a. Each motor driven pump develops a differential pressure of $\geq$ 1290 psid on recirculation flow of $\geq$ 100 gpm, and

b. The steam turbine driven pump develops a differential pressure of $\geq$ 1310 psid on recirculation flow of $\geq$ 220 gpm when the secondary steam pressure is greater than 600 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Reverifying the requirements of Tech Spec. surveillance 4.7.1.2.a.2 by a second and independent operator.

4. Establish and maintain constant communications between the control room and the auxiliary feed pump room while any normal discharge valve is closed during surveillance testing.

5. Verifying operability of each Service Water auxiliary supply valve by cycling each manual Service Water to Auxiliary Feedwater System valve through one complete cycle.

b. Following a plant outage of 30 days or greater, verify Auxiliary Feedwater Flow from TK-210 to the Steam Generators with the Auxiliary Feedwater Valves in their normal alignment.

c. At least once per 18 months during shutdown by:

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on an auxiliary feedwater actuation test signal.

3. Verifying that each auxiliary feedwater pump starts automatically upon receipt of an auxiliary feedwater actuation test signal.

PRIMARY PLANT DEMINERALIZED WATER (PPDW)

LIMITING CONDITION FOR OPERATION

3.7.1.3 The primary plant demineralized water storage tank shall be OPERABLE with a minimum contained volume of 127,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With less than 127,000 gallons of water in the PPDW storage tank, within 4 hours either:

a. Restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the auxiliary feedwater pumps and restore the PPDW storage tank water volume to within its limit within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The PPDW storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be ≤ 0.10 µCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific-activity of the secondary coolant system > 0.10 µCi/gram DOSE EQUIVALENT I 131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODES 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours;

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed after:

a. The inoperable isolation valve is restored to OPERABLE status, or

b. The isolation valve is maintained closed;

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve that is open shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 5 seconds on any automatic closure actuation signal while in HOT STANDBY with $T_{avg} \geq 515°F$ during each reactor shutdown except that verification of full closure within 5 seconds need not be determined more often than once per 92 days.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side ≤ 200 psig within 30 minutes, and

b. Perform an analysis to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant in the steam generator is < 70°F.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two primary component cooling water subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one less than two primary component cooling water subsystems OPERABLE, restore at least two subsystems to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two primary component cooling water subsystems shall be demonstrated OPERABLE.

a. Verify that each pump develops the required differential pressure and flow rate when tested in accordance with the requirements of Section 4.0.5.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown, by cycling each power operated valve servicing safety related equipment that is not testable during plant operation, through at least one complete cycle of full travel.

3/4.7.4 SERVICE WATER SYSTEM (SWS)

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two service water subsystems supplying safety related equipment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one less than two SWS subsystems OPERABLE, restore at least two subsystems to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two SWS subsystems shall be demonstrated OPERABLE:

a. Verify that each pump develops the required differential pressure and flow rate when tested in accordance with the requirements of Section 4.0.5.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown, by cycling each power operated valve servicing safety related equipment that is not testable during plant operation, through at least one complete cycle of full travel.

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3/4.7.5 ULTIMATE HEAT SINK - OHIO RIVER

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 654 Mean Sea Level, at the intake structure, and

b. An average water temperature of $\leq$ 86°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

PLANT SYSTEMS

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Ohio River exceeds 695 Mean Sea Level at the intake structure.

APPLICABILITY: At all times.

ACTION:

With the water level at the intake structure above elevation 695 Mean Sea Level:

a. Be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours, and

b. Initiate and complete within 8 hours, the following flood protection measures:

   1. Install and seal the flood doors in the intake structure.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level at the intake structure shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 690 Mean Sea Level, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 690 Mean Sea Level.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The Control Room Emergency Air Cleanup and Pressurization System comprised of the following shall be OPERABLE:

a. A pressurization filtration unit comprised of two trains of fans and filters, and flow path control dampers.**

b. A bottled air pressurization system comprised of 5 subsystems with two bottles in each subsystem.*

c. Two isolation dampers in series in each of four normal air flow paths (two intake and two exhaust) with each damper OPERABLE by automatic actuation or OPERABLE by being secured in a closed position with power removed.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With one train of the pressurization filtration unit, or one subsystem of the bottled air pressurization system, or one of two isolation dampers in series inoperable, restore the system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one train of the pressurization filtration unit or one subsystem of the bottled air pressurization system, or one of two isolation dampers in series inoperable, restore the inoperable system to OPERABLE status within 7 days or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel and movement of loads over irradiated fuel.

b. With both trains of the pressurization filtration unit, or more than one subsystem of the bottled air pressurization system, or two of two isolation dampers in series inoperable suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel and movement of loads over irradiated fuel.

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*The air bottles may be isolated for up to 8 hours for performance of instrumentation and control systems testing.

**Emergency backup power for one train of dampers and fans of the pressurization filtration unit not required in MODES 5 and 6.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

SURVEILLANCE REQUIREMENTS

4.7.7.1 The Control Room Emergency Air Cleanup and Pressurization System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is ≤ 88°F.

b. At least once per 31 days by:

   1. Initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for 15 minutes with the heaters in operation.

   2. Verifying that the bottled air pressurization system contains a minimum of 10 bottles of air each pressurized to at least 1825 psig and that each solenoid operated valve is powered from an operable emergency bus.

c. At least once per 18 months or (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings by:

   1. Verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the pressurization filtration system at a flow rate of 800 to 1000 cfm.

   2. Verifying that the HEPA filter banks remove ≥ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the pressurization filtration system at a flow rate of 800 to 1000 cfm.

   3. Verifying a system flow rate of 800 to 1000 cfm during system operation.

d. At least once per 18 months or (1) after 720 hours of system operation, or (2) following painting, fire or chemical release in the vicinity of control room outside air intakes while the systems is operating subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of ≥ 99% for radioactive methyl iodide at an air flow velocity of 0.7 ft/sec ± 20% with an inlet methyl iodide concentration of 1.5 to 2.0 mg/m^3, ≥ 70% relative humidity, and 30°C ± 1/2°C; other test conditions shall be in accordance with ANSI N510-1980. The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

e. At least once per 18 months by:

1. Verifying that the pressure drop for the combined HEPA filters and charcoal adsorber banks is less than 5.6 inches Water Gauge while operating the pressurization filtration system at a flow rate of 800 to 1000 cfm.

2. Verifying that on a Containment Isolation Phase B/Control Room High Radiation test signal, the system automatically closes all the series isolation ventilation system dampers which isolate the control room from the outside atmosphere and the system automatically starts 60 minutes later and supplies air to the control room through the HEPA filters and charcoal adsorber banks.

3. Verifying that on a chlorine test signal from the system automatically closes all the series isolation ventilation system dampers which isolate the combined control room from the outside atmosphere.

4. Verifying that the pressurization filtration system maintains the control room at a positive pressure of ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

5. Verifying that the heaters dissipate 5 ± 0.5 kw when tested in accordance with ANSI N510-1980.

6. Verifying that a chlorine/control room high radiation/containment phase B isolation signal will initiate operation of the bottled air pressurization system.

7. Verifying by a partial discharge test from four out of five subsystems of the bottled air pressurization system at a discharge flow of less than 1000 cfm that the bottled air pressurization system will pressurize the control room to ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two SLCRS exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one SLCRS exhaust air filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each SLCRS exhaust air filter train shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the "standby" HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes with the heater controls operational.

b. At least once per 18 months and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire or chemical release in any ventilation zone communicating with the system by:

  1. Verifying that the charcoal adsorbers remove $\geq$ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 59,000 cfm ± 10%.

  2. Verifying that the HEPA filter banks remove $\geq$ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 59,000 cfm ± 10%.

  3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of $\geq$ 99% for radioactive methyl iodide at an air flow velocity of 0.7 ft/sec ± 20% with an inlet methyl iodide concentration of 1.5 to 2.0. mg/m^3, $\geq$ 70% relative humidity, and 30°C ± ½°C; other test conditions shall be in accordance with ANSI N510-1980. The carbon samples not obtained from test canisters shall be taken with a slotted tube sampler in accordance with ANSI N509-1980.

SURVEILLANCE REQUIREMENTS

4. Verifying a system flow rate of 59,000 cfm ± 10% during system operation.

c. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.8 inches Water Gauge while operating the ventilation system at a flow rate of 59,000 cfm ± 10%.

2. Verifying that the exhaust from the contiguous area is diverted through the SLCRS filter train on a Containment Isolation - Phase "A" signal in less than 5 minutes.

d. Verifying that the air flow distribution to each HEPA filter and charcoal adsorber is within ± 20% of the averaged flow per unit after initial installation and after any maintenance affecting the flow distribution.

e. At least once per 4 months of system operation, perform the surveillance requirement of 4.7.8.1 b.3.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 Each sealed source containing radioactive material either in excess of those quantities of byproduct material listed in 10 CFR 30.71 or $\geq$ 0.1 microcuries of any other material, including alpha emitters, shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: AT ALL TIMES.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.9.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Sources in use (excluding startup sources previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive materials.

   1. With a half-life greater than 30 days (excluding Hydrogen 3) and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (CONTINUED)

b. Stored sources not in use - Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources - Each sealed startup source shall be tested prior to being subjected to core flux and following repair or maintenance to the source.

4.7.9.1.3 Reports - A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3/4.7.10 (This specification number is not used)

3/4.7.11 (This specification number is not used)

3/4.7.12 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.12 All snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems* required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.12.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.12 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after four months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period. ** #
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

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*These systems are defined as those portions or subsystems required to prevent releases in excess of 10 CFR 100 limits.

**The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (CONTINUED)

The snubbers may be categorized into two groups: those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

b. Visual Inspection Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.12.d. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (of at least 10%) of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For Functional Testing type of snubber shall mean a group or combination of groups by load size and kind (i.e., hydraulic or mechanical) or any other combination of load size and kind. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.12.d, an additional 10% shall be functionally tested.

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SURVEILLANCE REQUIREMENTS (Continued)

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle.
2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.).
3. Snubbers within 10 feet of the discharge from a safety relief valve.

Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.*

If a spare snubber has been installed in place of a failed snubber, the spare snubber shall be retested. Test results of this snubber may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

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*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

d. Snubber Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.
2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension.
3. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

e. Service Life Monitoring

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and may be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2. Service life will be defined to commence at plant startup subsequent to initial fuel load.

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SWE)

LIMITING CONDITION FOR OPERATION

3.7.13.1 At least one standby service water subsystem shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With less than one SWE subsystem OPERABLE, restore at least one subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following thirty hours.

SURVEILLANCE REQUIREMENTS

4.7.13.1 At least one SWE subsystem shall be demonstrated OPERABLE:

a. By verifying that each pump develops at least 109 psid differential pressure while pumping through its test flow line when tested pursuant to Specification 4.0.5.

b. At least once per 18 months during shutdown by starting a Standby Service Water System Pump, shutting down one Service Water System Pump, and verifying that the Standby Service Water Subsystem provides at least 8584 gpm cooling water to that portion of the Service Water System under test for at least 2 hours.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators each with:

1. Separate day tank containing a minimum of 350 gallons of fuel,
2. A separate fuel storage system containing a minimum of 53,225 gallons of fuel,
3. A separate fuel transfer pump,
4. Lubricating oil storage containing a minimum total volume of 504 gallons of lubricating oil, and
5. Capability to transfer lubricating oil from storage to the diesel generator unit.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.5 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.5 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in COLD SHUTDOWN within the next 36 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

LIMITING CONDITION FOR OPERATION (Continued)

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirements 4.8.1.1.2.a.5 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 4 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determine OPERABLE at least once per 7 days by verifying correct breaker alignment, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the unit circuit to the system circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the fuel storage tank,

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975 when checked for viscosity, water and sediment,

4. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

5. Verifying the diesel starts from ambient condition,

6. Verifying the generator is synchronized, loaded to $\geq$ 4,238 kw, and operates for at least 60 minutes, and

7. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

8. Verifying the lubricating oil inventory in storage.

b. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load of $\geq$ 825 kw without tripping,

3. Simulating a loss of offsite power in conjunction with a safety injection signal, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

4. Verifying that on a loss of power to the emergency busses, all diesel generator trips, except engine overspeed, generator differential current, and generator overexcitation are automatically disabled.

5. Verifying the diesel generator operates for at least 60 minutes while loaded to $\geq$ 4,238 kw.

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 4,535 kw.

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within ±10% of its required value.

c. Check for and remove accumulated water:

1. From the day tank, at least once per 31 days and after each operation of the diesel where the period of operation was greater than 1 hour, and

2. From the fuel oil storage tank, at least once per 92 days.

d. At least once per 92 days and from new fuel oil prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM D270-1975 meets the following minimum requirements and is tested within the specified time limits:

1. As soon as a sample is taken (or prior to adding new fuel to the storage tank) verify in accordance with the tests specified in ASTM D975-1977 that the sample has:

a) A water and sediment content of less than or equal to 0.05 volume percent;

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes;

c) An API Gravity of within 0.3 degrees of 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API Gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees; and

2. Within one week after obtaining the sample, verify an impurity level of less than 2 milligrams of insolubles per 100 milliliter is met when tested in accordance with ASTM D2274-1970. The analysis on the sample may be performed after the addition of new fuel oil.

3. Within two weeks of obtaining the sample, verify that the other properties specified in Table 1 of ASTM D975-1977 and Regulatory Guide 1.137 Position 2.a are met (when tested in accordance with ASTM D975-1977). An analysis for sulfur shall be performed in accordance with ASTM D129, ASTM D1552-1979 or ASTM D2622-1982.

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 514 rpm in less than or equal to 10 seconds.

f. At least once per 10 years by:

   1) Draining each main fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite solution or other appropriate cleaning solution, and

   2) Performing a pressure test, of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code, at a test pressure equal to 110% of the system design pressure.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

    1. Day tank containing a minimum of 350 gallons of fuel,

    2. A fuel storage system containing a minimum of 53,225 gallons of fuel,

    3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.6.

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEM

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators with tie breakers open between redundant busses:

4160 volt Emergency Bus #2AE and 480V Emergency Bus #2N

4160 volt Emergency Bus #2DF and 480V Emergency Bus #2P

120 volt A.C. Vital Bus #I

120 volt A.C. Vital Bus #II

120 volt A.C. Vital Bus #III

120 volt A.C. Vital Bus #IV

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated power availability.

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A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator.

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Bus

2 - 120 volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be energized and OPERABLE:

TRAIN "A" (orange) consisting of 125-volt D.C. busses No. 2-1 & 2-3, 125-volt D.C. battery banks 2-1 & 2-3 & charger 2-1 and rectifier 2-3.

TRAIN "B" (purple) consisting of 125-volt D.C. busses No. 2-2 & 2-4, 125-volt D.C. battery banks 2-2 & 2-4 and charger 2-2 and rectifier 2-4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers or rectifiers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.3.2.a.1 within one hour. Within 4 hours place in service spare charger 2-7 or continue the Surveillance Requirement of 4.8.2.3.2.a.1 at least once per 8 hours thereafter. If any Category A limit in Table 3.8-1 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.3.2 Each 125-volt battery bank and charger/rectifier shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 3.8-1 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 127.8 - volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 -volts, or battery overcharge with battery terminal voltage above 150 -volts, by verifying that:

1. The parameters in Table 3.8-1 meet the Category B limits.
2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohms, and
3. The average electrolyte temperature of every tenth cell of connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,
2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,
3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x 10^{-6} ohms; and
4. The battery charger will supply at least 100 amperes at 140-volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the 2-hour design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. At least once per 18 months, during shutdown, performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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TABLE 3.8-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	$\geq$ 1.200(b)	$\geq$ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than (2) amps when on charge.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION OF OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

2 - 125-volt D.C. bus systems, and

2 - 125-volt battery bank and chargers/rectifiers associated with the above D.C. bus systems.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus system OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus system shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt battery bank and chargers/rectifiers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

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3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

Additionally, valve 2CHS-91 shall be closed and secured in position.

APPLICABILITY: MODE 6*

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 30 gpm of greater than or equal to 7000 ppm boric acid solution or its equivalent until K_{eff} is reduced to ≤ 0.95 or the boron concentration is restored to $\geq$ 2000 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

b. With valve 2CHS-91 not closed and secure in position, immediately close and secure in position.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head. and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

4.9.1.3 Valve 2CHS-91 shall be verified closed and locked secure in position at least once per 31 days.

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*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

INSTRUMENTATION

LIMITING CONDITION OF OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

DECAY TIME

LIMITING CONDITION OF OPERATION

3.9.3 The reactor shall be subcritical for at least 150 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel

ACTION:

With the reactor subcritical for less than 150 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 150 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Exhausting at less than or equal to 7500 cfm through OPERABLE Containment Purge and Exhaust Isolation Valves with isolation times as specified in Table 3.6-1 to OPERABLE HEPA filters and charcoal adsorbers of the Supplemental Leak Collection and Release System (SLCRS).

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required condition within 150 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment.

4.9.4.2 The containment purge and exhaust system shall be demonstrated OPERABLE by:

a. Verifying the flow rate to the SLCRS at least once per 24 hours when the system is in operation.

b. Testing the Containment Purge and Exhaust Isolation Valves per the applicable portions of Specification 4.6.3.1.2, and

c. Testing the SLCRS per Specification 4.7.8.1 with the exception of item 4.7.8.1.c.2.

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

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MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

  1. A minimum capacity of 3250 pounds, and

  2. An overload cut off limit $\leq$ 2700 pounds.

b. The auxiliary hoist used for movement of control rods having:

  1. A minimum capacity of 700 pounds, and

  2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 150 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2700 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 150 hours prior to the start of such operations by performing a load test of at least 700 pounds.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 3000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 3000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

3/4 9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The residual heat removal loop may be removed from operation for up to 4 hours per 8 hour period during the performance of Ultrasonic In-service Inspection inside the reactor vessel nozzles provided there is at least 23 feet of water above the top of the reactor vessel flange.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm at least once per 4 hours when making boron dilution changes and ≥ 1000 gpm for decay heat removal when the Reactor Coolant System is in the drained down condition within the loops.

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*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 2-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per specification 4.0.5.

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*The normal or emergency power source may be inoperable for each RHR loop.

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 150 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation signal from each of the containment radiation monitoring instrumentation channels.

3/4 9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

FUEL BUILDING VENTILATION SYSTEM - FUEL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.9.12 The fuel building portion of the Supplemental Leak Collection and Release System (SLCRS) shall be operating and discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers during either:

a. Fuel movement within the spent fuel pool, or

b. Crane operation with loads over the spent fuel storage pool.

APPLICABILITY: When irradiated fuel which was decayed less than 60 days is in the fuel storage pool.

ACTION:

With the requirement of the above specification not satisfied, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The fuel building portion of the SLCRS shall be verified to be operating with all building doors closed within 2 hours prior to the initiation of and at least once per 12 hours during either fuel movement within the fuel storage pool or crane operation with loads over the fuel storage pool.

FUEL BUILDING VENTILATION SYSTEM - FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 The fuel building portion of the Supplemental Leak Collection and Release System (SLCRS) shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

Without the fuel building portion of the SLCRS OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until the SLCRS portion of the fuel building ventilation system is restored to OPERABLE status. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 The fuel building portion of the SCLRS shall be demonstrated OPERABLE by testing the SLCRS per Specification 4.7.8.

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3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirements of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2

ACTION:

a. With the reactor critical ($K_{eff} \geq 1.0$) and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 30 gpm of $\geq$ 7000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 30 gpm of $\geq$ 7000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained $\leq$ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either;

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be $\leq$ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

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SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.1.5, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at $\leq$ 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is $\geq$ 531°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) < 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 531°F at least once per 30 minutes during PHYSICS TESTS.

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.2 may be suspended during the performance of hot rod drop time measurements in MODE 3 provided at least two reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During performance of hot rod drop time measurements.

ACTION:

> With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately open the reactor trip breakers and comply with the provisions of the ACTION statements of Specification 3.4.1.2.

SURVEILLANCE REQUIREMENTS

4.10.4 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability and by verifying secondary side narrow range water level to be greater than or equal to 15.5 %.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released at anytime from the site (See Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table 11, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} µCi/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released from the site to unrestricted areas exceeding the above limits; immediately restore concentration within the above limits, and

b. Submit a Special Report to the Commission within 30 days in accordance with Specification 6.9.2.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1*.

4.11.1.1.2 The results of radioactive analysis shall be used in accordance with the methods of the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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*Radioactive liquid discharges are normally via batch modes. Turbine Building Drains shall be monitored as specified in Section 4.11.1.1.3. Recirculation drain pump discharge shall be monitored as specified in Section 4.11.1.1.4.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.1.1.3 When the activity of the secondary coolant is greater than 10^{-5} μCi/ml gross and the Turbine Building transfer pumps (2DBS-P42, 2DBS-P43, 2DBS-P44) are not pumping their sumps to the steam generator blowdown tank (2SGC-TK21B), grab samples shall be taken for each sump discharge from the turbine building. The sample shall be analyzed for gross activity at a sensitivity of at least 10^{-7} μCi/ml and recorded in plant records. Water volume discharged shall be estimated from the number of pump operations unless alternate flow or volume instrumentation is provided.

4.11.1.1.4 Prior to the Recirculation Drain Pump(s) (2DAS-P215A/B) discharging to catch basin 16, a grab sample will be taken. The samples will be analyzed for gross activity at a sensitivity of at least 10^{-7} μCi/ml. Water volume discharged shall be estimated from the number of pump operations unless alternate flow or volume instrumentation is provided.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch[h]	P Each Batch[h]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M[h]	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch[h]	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch[h]	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e, g]	Grab Sample[g]	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	Grab Sample[g]	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Grab Sample[g]	M Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Grab Sample[g]	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$\mathrm{LLD} = \frac{4.66\ s_b}{(E)\ (V)\ (2.22)\ (Y)\ \exp(-\lambda\Delta T)}$$

where:

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume);

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute);

E is the counting efficiency (as counts per transformation);

V is the sample size (in units of mass or volume);

2.22 is the number of transformations per minute per picocurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide;

ΔT is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium in milk samples). Typical values of E, V, Y and ΔT should be used in the calculations.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

TABLE 4.11-1 (Continued)

TABLE NOTATION

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release exists when the discharge of liquid wastes is from a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

e. A continuous release exists when the discharge of liquid wastes is from a nondiscrete volume; e.g., from a volume of a system having an input flow during the continuous release. This is applicable to the Turbine Building drains when the secondary coolant gross radioactivity (beta and gamma) is greater than 10^{-5} µCi/ml.

f. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should be reported as "less than" the nuclide's LLD, and should not be reported as being present at the LLD level for that nuclide. The "less than" values should not be used in the required dose calculations. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the semi-annual Radioactive Effluent Release Report.

g. Whenever there is primary to secondary leakage, sampling is done for turbine building drain effluents by means of grab samples taken every 4 hours during the period of discharge and analyzed for gross radioactivity (beta and gamma) at a sensitivity of at least 10^{-7} µCi/ml and recorded in the plant records, along with the flow rate. Primary to secondary leakage is considered to be occurring whenever measurements indicate that secondary coolant gross radioactivity (beta and gamma) is greater than 10^{-5} µCi/ml. In addition, two (2) plant personnel shall check release calculations to verify that the limits of 3.11.1.1 and 3.11.1.2 are not exceeded.

h. Whenever the Recirculation Drain Pump(s) are discharging to catch basin 16, sampling will be performed by means of a grab sample taken every 4 hours during pump operation.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to MEMBER(S) OF THE PUBLIC from radioactive materials in liquid effluents released from the reactor unit (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases, and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits. (This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act).*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable

SURVEILLANCE REQUIREMENTS

4.11.1.2.1 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

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*Applicable only if drinking water supply is taken from the receiving water body.

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be used to reduce the radioactive materials in each liquid waste batch prior to its discharge when the projected doses due to liquid effluent releases from the reactor unit (See Figure 5.1-2) when averaged over 31 days would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With liquid waste being discharged without treatment and exceeding the limits specified, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

    1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

    2. Action(s) taken to restore the inoperable equipment to operational status, and

    3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each miscellaneous temporary outside radioactive liquid storage tank shall be limited to $\leq$ 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit, and

b. Submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2 and include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

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RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in the unrestricted areas (see Figure 5.1-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be $\le$ 500 mrem/yr to the total body and $\le$ 3000 mrem/yr to the skin*, and

b. The dose rate limit, inhalation pathway only, for I-131, tritium and all radionuclides in particulate form (excluding C-14) with half-lives greater than 8 days shall be $\le$ 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately decrease the release rate to comply with the above limit(s), and

b. Submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate, inhalation pathway only, for I-131, tritium and all radionuclides in particulate form (excluding C-14) with half-lives greater than 8 days in gaseous effluents, shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

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*During containment purge the dose rate may be averaged over 960 minutes.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
B. Containment Purge	P Each Purge[b] Grab Sample	P Each Purge[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. Ventilation Systems[h] 1. Process Vent 2. Elevated Release Point 3. Ventilation Vent 4. Aux. Bldg. Exh. Hood (Emergency Use Only) 5. Cond. Polish. Bldg. Vent. Exh. 6. Decon. Bldg. Vent. Exh. 7. Waste Gas Storage Vault Vent. Exh. 8. Turbine Building Exhaust	M[b,c,e,] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}

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TABLE 4.11-2 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
D. All systems listed above which produce continuous release.	Continuous[f]	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f]	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The Lower Limit of Detection (LLD) is defined in Table Notation (a) of Table 4.11-1 of Specification 4.11.1.1.

b. Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period. This requirement does not apply if (1) analysis shows that the Dose Equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

c. Tritium grab samples shall be taken at least once per 24 hours (from the appropriate ventilation release path) when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The average ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specification 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the semi-annual effluent report.

h. Only when this release path is in use.

RADIOACTIVE EFFLUENTS

DOSE-NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose from the reactor unit in unrestricted areas (See Figure 5.1-1) due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to $\leq$ 5 mrad for gamma radiation and $\leq$ 10 mrad for beta radiation.

b. During any calendar year, to $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions shall be determined in accordance with the ODCM at least once every 31 days.

DOSE-RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to MEMBER(S) OF THE PUBLIC from radioiodines and radioactive materials in particulate form (excluding C-14), and radionuclides (other than noble gases) with half-lives greater than 8 days in gaseous effluents released from the reactor unit (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter to ≤ 7.5 mrem to any organ, and

b. During any calendar year to ≤ 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioiodines, radioactive materials in particulate form, (excluding C-14), and radionuclides (other than noble gases) with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations. Cumulative dose contributions shall be determined in accordance with the ODCM at least once every 31 days.

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GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the reactor unit (see Figure 5.1-1), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the reactor unit (see Figure 5.1-1) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information.

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to operational status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

GASEOUS WASTE STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.5 The quantity of radioactivity contained in any connected group of gaseous waste storage tanks shall be limited to $\leq$ 19,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any connected group of gaseous waste storage tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tanks and within 48 hours reduce the tanks contents to within the limit, and

b. Submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2 and include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 The quantity of radioactive material contained in any connected group of gaseous waste storage tanks shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tanks.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of oxygen in the waste gas holdup system shall be limited to $\leq$ 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system > 2% by volume, immediately suspend all additions of waste gases to the gaseous waste decay tank and reduce the concentration of oxygen to $\leq$ 4% within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the affected tank and reduce the concentration of oxygen to less than or equal to 2% by volume within twelve hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentrations of oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10 or monitoring in conjunction with its associated action statement.

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3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3.1 The solid radwaste system shall be used, as applicable, to solidify and package radioactive wastes, and to ensure meeting the requirements of 10 CFR Part 20, 10 CFR Part 61 and of 10 CFR Part 71. Methods utilized to meet these requirements shall be described in facility procedures and in the Process Control Program (PCP).

APPLICABILITY: At all times.

ACTION:

a. With the applicable requirements of 10 CFR Part 20, 10 CFR Part 61 and 10 CFR Part 71 not satisfied, suspend affected shipments of solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1.1 Prior to shipment, solidification shall be verified in accordance with Station Operating Procedures.

4.11.3.1.2 Reports. The semi-annual Radioactive Effluent Release Report in Specification 6.9.1.12 shall include the following information for each type of solid waste shipped offsite during the report period:

a. container volume;

b. total curie quantity (determined by measurement or estimate);

c. principal radionuclides (determined by measurement or estimate);

d. type of waste (e.g., spent resin, compacted dry waste evaporator bottoms);

e. type of container (e.g., LSA, Type A, Type B, Large Quantity);

f. solidification agent (e.g., cement); and

g. classification and other requirements specified by 10 CFR Part 61.

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4.1 The dose or dose commitment to MEMBER(S) OF THE PUBLIC from all facility releases is limited to $\leq$ 25 mrem to the total body or any organ (except the thyroid, which is limited to $\leq$ 75 mrem) for a calendar year.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, prepare and submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2 defining the corrective action and limit the subsequent releases such that the dose or dose commitment to MEMBER(S) OF THE PUBLIC is limited to $\leq$ 25 mrem to the total body or any organ (except thyroid, which is limited to $\leq$ 75 mrem) for a calendar year. This special report shall describe the steps to be taken or modifications necessary to prevent a recurrence. Otherwise, obtain a variance from the Commission to permit releases which exceed the 40 CFR 190 Standard.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Dose Calculations. Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, and 3.11.2.3.b and in accordance with the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

b. With the level of radioactivity in an environmental sampling medium at one or more of the locations specified in Table 3.12-1 exceeding the limits of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of affected calendar quarter a Special Report pursuant to Specification 6.9.2 which includes an evaluation of any release conditions, environmental factors or other aspects which caused the limits of Table 3.12-2 to be exceeded This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Report.

When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Limit Level (1)}} + \frac{\text{Concentration (2)}}{\text{Limit Level (2)}} + \ldots \geq 1.0$$

c. With milk or fresh leafy vegetable samples unavailable from the required number of locations selected in accordance with Specification 3.12.2 and listed in the ODCM, obtain replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1 and the ODCM provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations, if available.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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3/4.12 RADIOLIGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM (Continued)

SURVEILLANCE REQUIREMENTS

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type and Frequency(a) of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	5 locations 1. One sample from a control location 10-20 miles distant and in the least prevalent wind direction. 2. One sample from vicinity of community having the highest calculated annual average ground level D/Q.	Continuous operation of sampler with sample collection at least weekly.	Each radioiodine canister. Analyze for I-131; Particulate sampler. Analyze for gross beta weekly(b); Perform gamma isotopic analysis on composite (by location) sample at least quarterly.
2. DIRECT RADIATION	40 Locations. ≥2 TLD or a pressurized ion chamber at each location.	Continuous measurement with collection at least quarterly.	Gamma dose, quarterly.

(a) Analysis frequency same as sampling frequency unless otherwise specified.

(b) Particulate samples are not counted for >24 hours after filter change. Perform gamma isotopic analysis on each sample when gross beta is >10 times the yearly mean of control samples.

**Sample locations are given on figures and table in Offsite Dose Calculation Manual (ODCM).

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type and Frequency[(a)] of Analysis
3. WATERBORNE			
a. Surface	2 Locations. 1. One sample upstream. 2. One sample downstream.	Composite* sample collected over a period not to exceed one month.	Gamma isotopic analysis of each composite sample; Tritium analysis of composite sample at least quarterly.
b. Drinking	2 Locations.	Composite* sample collected over a period not to exceed 2 weeks.	I-131 analysis of each composite sample; Gamma isotopic analysis of composite sample (by location) monthly; Tritium analysis of composite sample quarterly.
c. Groundwater	N/A - No wells in lower elevations between plant and river.		
d. Sediment from Shoreline	1 Location	Semi-Annually	Gamma isotopic analysis semiannually.

(a) Analysis frequency same as sampling frequency unless otherwise specified.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.

**Sample locations are given on figures and table in Offsite Dose Calculation Manual (ODCM).

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type and Frequency[(a)] of Analysis
4. INGESTION			
a. Milk	4 Locations.[(c)] 1. Three samples selected on basis of highest potential thyroid dose using milch census data. 2. One local large dairy.	At least bi-weekly when animals are on pasture; at least monthly at other times.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	2 Locations.	Semi-Annual. One sample of available species.	Gamma isotopic analysis on edible portions.
c. Food Products (Leafy Vegetables)	4 Locations. 1. Three locations within 5 miles. 2. One control location.	Annually at time of harvest.	Gamma isostopic analysis and I-131 analysis on edible portion.

(a) Analysis frequency same as sampling frequency unless otherwise specified.

(c) Other dairies may be included as control station or for historical continuity. These would not be modified on bsis of milch animal census.

**Sample locations are given on figures and table in Offsite Dose Calculation Manual.

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TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

	Reporting Levels				
Analysis	Water pCi/l	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Broad Leaf Vegetables (pCi/Kg, wet)
H-3	2×10^4 (a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
CS-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2			3×10^2	

(a) For drinking water samples. This is a 40 CFR Part 141 value.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a]

Analysis	Water pCi/l	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-95	30^c					
Nb-95	15^c					
I-131	1^b	7×10^{-2}		1	60	
CS-134	15	5×10^{-2}	130	15	60	150
Cs-137	18	6×10^{-2}	150	18	80	180
Ba-140	60^c			60		
La-140	15^c			15		

NOTE: This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall be identified and reported.

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$\text{LLD} = \frac{4.66\ S_b}{(E)\ (V)\ (2.22)\ (Y)\ \exp(-\lambda\Delta T)}$$

where:

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume);

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute);

E is the counting efficiency (as counts per transformation);

V is the sample size (in units of mass or volume);

2.22 is the number of transformations per minute per picocurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide;

ΔT is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of S_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and ΔT should be used in the calculations.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

b. LLD for drinking water.

c. If parent and daughter are totaled, the most restrictive LLD should be applied.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall indentify the location of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. For elevated releases as defined in Regulatory Guide 1.111, (Rev. 1) July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, which identifies the new location(s).

b. With a land use census identifying a milk animal location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1 prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days, if possible. The milk sampling program shall include samples from the three active milk animal locations, having the highest calculated dose or dose commitment. Any replaced location may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

c. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2.1 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey**, aerial survey, or by consulting local agriculture authorities.

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*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

**Confirmation by telephone is equivalent to door-to-door.

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Report.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3.1 The results of analyses performed as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Report.

BASES

FOR

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Condition for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.1 calls for each Reactor Coolant System accumulator to be OPERABLE and provides explicit ACTION requirements if one accumulator is inoperable. Under the terms of Specification 3.0.3., if more than one accumulator is inoperable, the unit is required to be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Quench Spray Subsystems, to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable: Under the terms of Specification 3.0.3., if both of the required Containment Quench Spray Subsystems are inoperable, the unit is required to be in at least HOT STANDBY within 6 hours, in a least HOT SHUTDOWN within the following 6 hours and in at least COLD SHUTDOWN in the next 24 hours. It is assumed that the unit is brought to the required MODE within the required times by promptly initiating and carrying out the appropriate ACTION statement.

3.0.4 This specification provides that entry into an OPERABLE MODE, or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.0.5 This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the ACTION statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.8.1.1 requires in part that two emergency diesel generators be OPERABLE. The ACTION statement provides for a 72 hour out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all system subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components, and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e, be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, action is required in accordance with this specification.

As a further example, Specification 3.8.1.1 requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. The ACTION statement provides a 24-hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits, would also be inoperable. This would dictate invoking the applicable ACTION statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, action is required in accordance with this specification.

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APPLICABILITY

BASES

In MODES 5 or 6 Specification 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg} The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. With $T_{avg} \leq 200°F$, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

The purpose of borating to the COLD SHUTDOWN boron concentration prior to blocking safety injection is to preclude a return to criticality should a steam line break occur during plant heatup or cooldown.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9370 cubic feet in approximately 30 minutes. The reactivity change rate associate with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the pressurizer is capable of being in an OPERABLE status with a steam bubble, 3) the reactor pressure vessel is above its minimum RT_{NDT} temperature and 4) the protective instrumentation is within its normal operating range.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are provided to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

With the RCS average temperature less than 200°F, Low Head Safety Injection pump may be used in lieu of the operable charging pump with a minimum open RCS vent of 3.14 square inches. This will provide latitude for maintenance and ISI examinations on the charging system for repair or corrective action and will ensure that boration and makeup are available when the charging pumps are out-of-service. An open vent insures that RCS pressure will not exceed the shutoff head of the Low Head Safety Injection pumps.

2SIS-MOV8888A and B are the Low Head Safety Injection Pump discharge isolation valves to the RCS cold legs, the valves must be closed prior to reducing RCS pressure below the RWST head pressure to prevent draining into the RCS. Emergency backup power is not required since these valves are outside containment and can be manually operated if required, this will allow the associated diesel generator to be taken out of service for maintenance and testing.

The technical specification limit on the refueling water storage tank has been established at 859,248 gallons to account for reactivity considerations and the NPSH requirements of the ECCS system and the water required for containment spray operation.

3/4.1.2 BORATION SYSTEMS (Continued)

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area > 3.0 ft^2) assuming complete mixing of the RWST, RCS, ECCS, chemical addition tank, containment spray system piping, and other water volumes that may eventually reside in the sump Post-LOCA with all control rods assumed to be out (ARO), 4) long term subcriticality following a steamline break assuming ARI-1 and to preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The limitations for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Substituting a Low Head Safety Injection pump for a charging pump in MODES 5 and 6 will not increase the probability of an overpressure event since the shutoff head of the Low Head Safety Injection pumps is below the setpoint of the overpressure protection system.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.77% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirements occur at EOL from full power equilibrium xenon conditions and requires 13,390 gallons of 7000 ppm borated water from the boric acid storage tanks or 58,965 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 350°F, one boron injection flow path is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.1.2 BORATION SYSTEMS (Continued)

The boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2315 gallons of 7000 ppm borated water from the boric acid storage tanks or 10,196 gallons of 2000 ppm borated water from the refueling water storage tank.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that 1) acceptable power distribution limits are maintained, 2) the minimum SHUTDOWN MARGIN is maintained, and 3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the movable control assemblies is established by observing rod motion and determining that rods are positioned within ± 12 steps (indicated position), of the respective group demand counter position. The OPERABILITY of the control rod position indication system is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Continuous monitoring of rod position with respect to insertion limits and rod deviation is provided by the rod insertion limit monitor and rod deviation monitor, respectively. If the rod deviation monitor or the rod insertion limit monitor is inoperable, the frequency of manual comparison of indicated rod position is increased to an interval of at least once per 4 hours.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core > 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$	Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.
$F^N_{\Delta H}$	Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ± 7% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time

AXIAL FLUX DIFFERENCE (AFD) (Continued)

duration limit of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% of RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS

$F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

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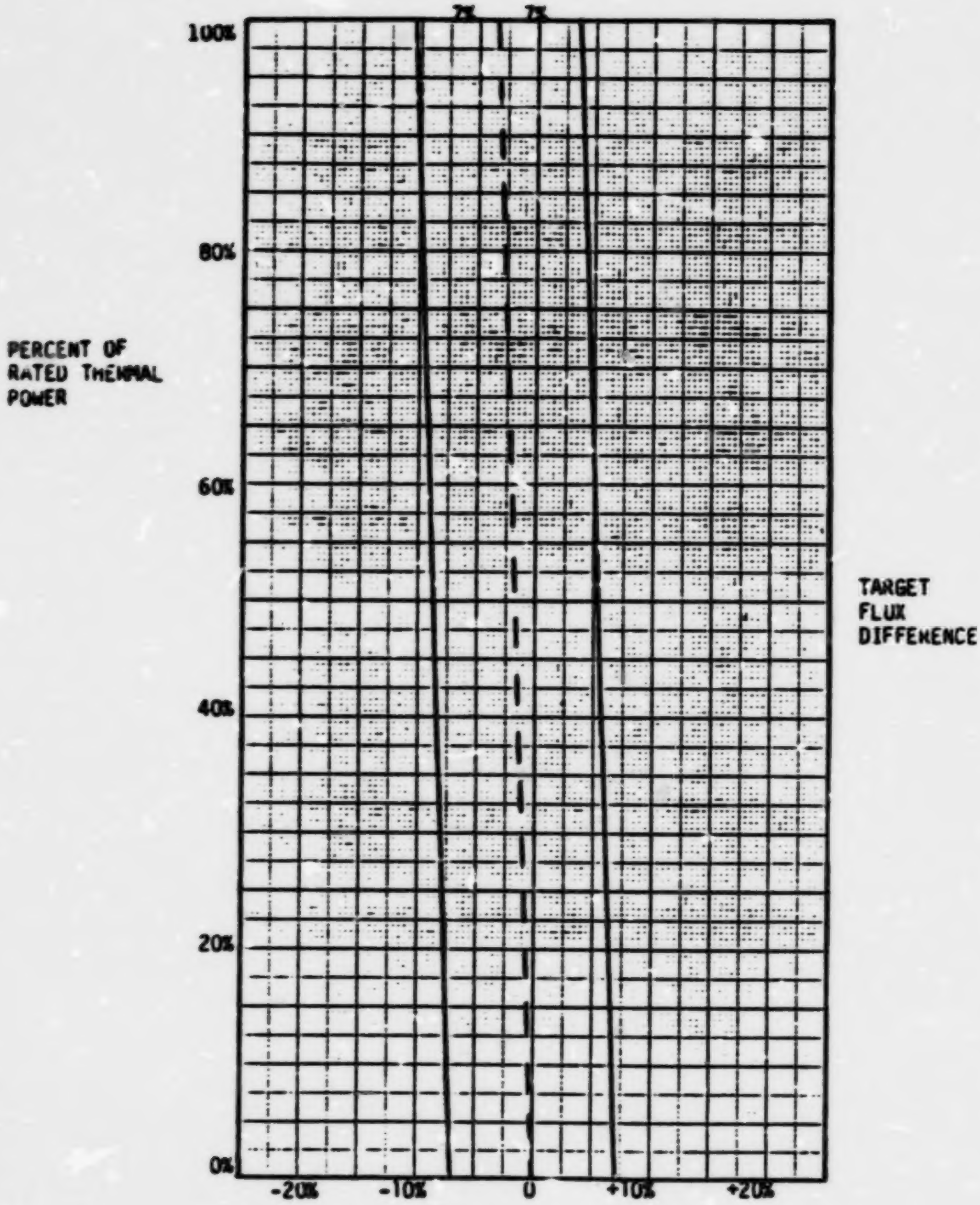

Figure B 3/4 2-1
TYPICAL INDICATED AXIAL FLUX DIFFERENCE (AFD)
VERSUS THERMAL POWER AT BOL

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3/4.2.2 and 3/4 2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS $F_Q(Z)$ AND $F^N_{\Delta H}$ (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate experimental error allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit of $F^N_{\Delta H}$ contains an 8% allowance for uncertainties which means that normal, full power, three loop operation will result in $F^N_{\Delta H} \leq 1.55/1.08$.

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, and completely offsets any rod bow penalties (< 3% for the worst case which occurs at a burnup of 33,000 MWD/MTU).

This margin includes the following:

1. Design Limit DNBR of 1.30 vs. 1.28
2. Grid Spacing (K_s) of 0.046 vs. 0.059
3. Thermal Diffusion Coefficient of 0.038 vs. 0.059
4. DNBR Multiplier of 0.865 vs. 0.88
5. Pitch reduction

The radial peaking factor F_{xy} (Z) is measured periodically to provide assurance that the hot channel factor, F_Q (Z), remains within its limit. The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.14 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The Quadrant Power Tilt Ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis.

3/4.2.4 QUADRANT POWER TILT RATIO (Continued)

Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainly on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$,

the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured' deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The surveillance requirements for the Manual Trip Function, Reactor Trip Breakers, and Reactor Trip Bypass Breakers are provided to reduce the possibility of an Anticipated Transient Without Scram (ATWS) event by ensuring OPERABILITY of the diverse trip features (Reference: Generic Letter 85-09).

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes # and ##) are based on values assumed on the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times (Note *), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and Non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

The maximum response time for control room isolation on high radiation is based on ensuring that the control room remains habitable following a small line break outside the containment. From a control room habitability aspect, the worst case accident that does not initiate a Containment Isolation - Phase B signal is the small line break outside the containment. This response time includes radiation monitor processing delays associated with the monitor averaging techniques. Diesel Generator starting and sequence loading delays are not included since these delays occur prior to the control room environment exceeding the high radiation setpoint.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Feature Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

P-11 Above the setpoint, P-11 automatically reinstates safety injection actuation on low pressurizer pressure, automatically blocks steamline isolation on high steam pressure rate, and enables safety injection and steamline isolation (with Loop Stop Valve Open) on low steamline pressure. Below the setpoint, P-11 allows the manual block of safety injection actuation on low pressurizer pressure, allows manual block of safety injection and steamline isolation (with Loop Stop Valve Open) on Low steamline pressure and enables steamline isolation on high steam pressure rate.

P-12 Above the setpoint, P-12 automatically reinstates an arming signal to the steam dump system. Below the setpoint P-12 blocks steam dump and allows manual bypass of the steam dump block to cooldown condenser dump valves.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: 1) the radiation levels are continually measured in the areas served by the individual channels; 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and 3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," October, 1980.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$, a full incore flux map is used.

Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the Quadrant Power Tilt Ratio when one Power Range Channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility and is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes."

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs."

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 (This Specification number is not used).

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures of ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2, with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event when the rod control system is capable of control bank rod withdrawal.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The requirement that 150 kw of pressurizer heaters and their associated controls and emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate

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3/4.4.5 STEAM GENERATORS (Continued)

decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a is 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, test, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 28 GPM with the modulating valve in the supply line fully open at RCS pressures in excess of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

3/4.4.6.3 PRESSURE ISOLATION VALVE LEAKAGE

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are

3/4.4.6.3 PRESSURE ISOLATION VALVE LEAKAGE (Continued)

important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of 10 CFR Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 µCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval

3/4.4.8 SPECIFIC ACTIVITY (Continued)

or exceeding the limits shown on Figure 3.4-1 must be restricted since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 10 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E >1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and phosphorus content of the material in question, can be predicted using Figures B 3/4.4-1 and Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damages to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} as well as adjustments for possible errors in the pressure and temperature sensing instruments. Additionally, these curves are not impacted by the special 10 CFR Part 50 rules for closure flange regions due to the low initial RT_{NDT} of the flange material.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nilductility temperature). The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (T_{NDT}) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and phosphorus) present in reactor vessel steels. The Regulatory Guide 1.99 Revision 1 curves which show the effect of fluence, copper content and phosphorus content on ΔRT_{NDT} for reactor vessel steels are shown in Figure B 3/4.4-2.

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TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS DATA

COMPONENT	CODE NO.	MATERIAL SPEC. NO.	Cu %	P %	T_{NDT} °F	50 FT/LB 35 MIL TEMP °F	RT_{NDT} °F	USE FT-LBS.
Closure Head Dome	B9008-1	A533B, CL.1	.13	.013	-20	50	-10	137
Closure Head Flange	B9002-1	A508, CL.2	---	.012	-10	<40	-10	136
Vessel Flange	B9001-1	A508, CL.2	---	.010	0	<10	0	132.5
Inlet Nozzle	B9011-1	A508, CL.2	---	.006	0	<10	0	104
Inlet Nozzle	B9011-2	A508, CL.2	---	.010	10	<10	10	115
Inlet Nozzle	B9011-3	A508, CL.2	---	.009	20	<40	20	122
Outlet Nozzle	B9012-1	A508, CL.2	---	.007	-10	< 0	-10	137
Outlet Nozzle	B9012-2	A508, CL.2	---	.006	-10	< 0	-10	121
Outlet Nozzle	B9012-3	A508, CL.2	---	.008	-10	< 0	-10	112
Nozzle Shell	B9003-1	A533B, CL.1	.13	.008	-10	110	50	91
Nozzle Shell	B9003-2	A533B, CL.1	.12	.009	0	120	60	79.5
Nozzle Shell	B9003-3	A533B, CL.1	.13	.008	-10	110	50	97.5
Inter. Shell	B9004-1	A533B, CL.1	.07	.010	0	120	60	83
Inter. Shell	B9004-2	A533B. CL.1	.07	.007	-10	100	40	75.5
Lower Shell	B9005-1	A533B, CL.1	.08	.009	-50	88	28	82
Lower Shell	B9005-2	A533B, CL.1	.07	.009	-40	93	33	77.5
Bottom Head Torus	B9010-1	A533B, CL.1	.15	.007	-30	56	-4	97
Bottom Head Dome	B9009-1	A533B, CL.1	.14	.007	-30	35	-25	116
Weld (Inter. & Lower Shell Long. Seams & Girth Seam)*			.08	.008	-30	<30	-30	144.5
HAZ (Plate B9004-2)			---	----	-80	40	-20	76

*Same heat of wire and lot of flux used in all seams including surveillance weldment.

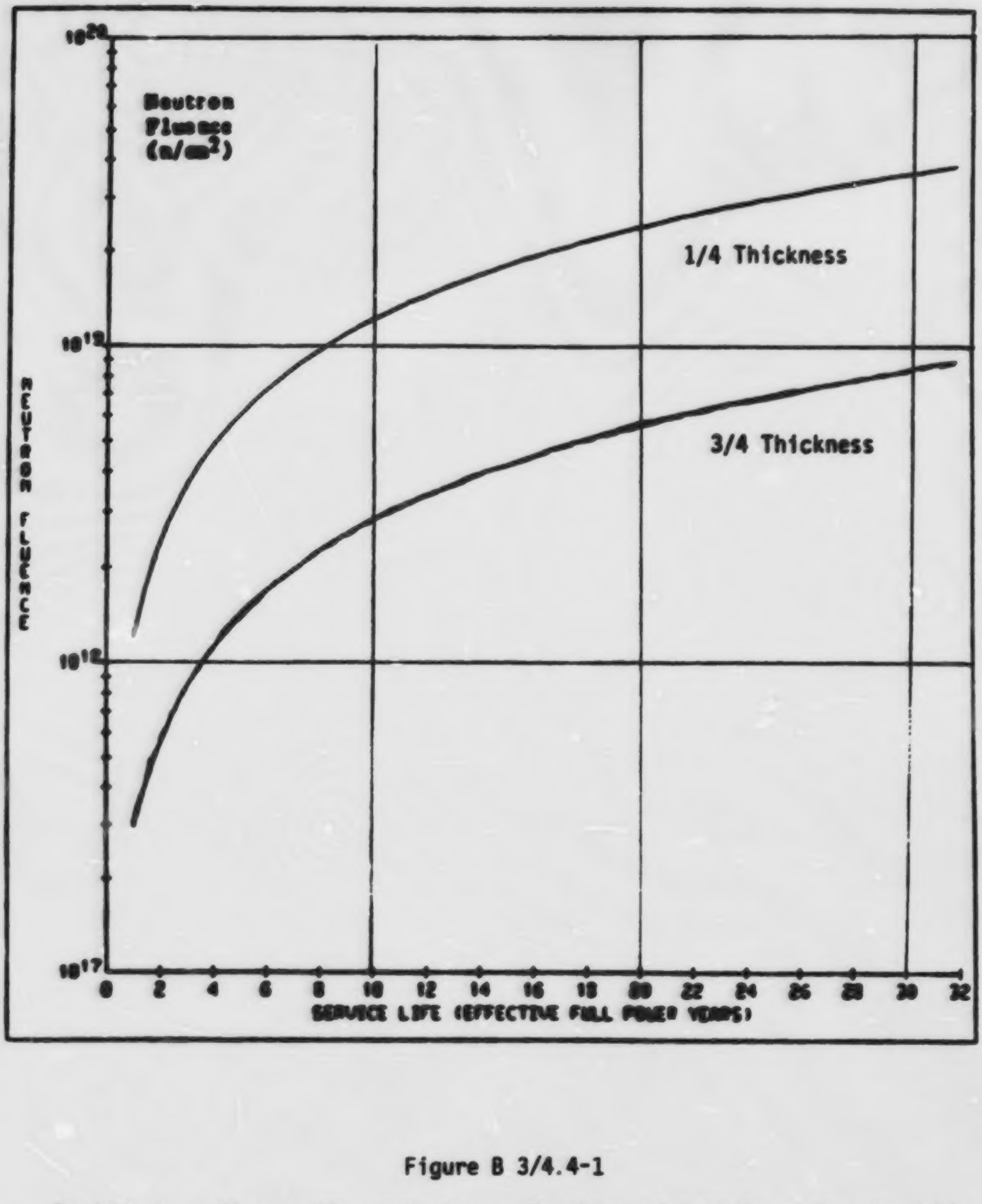

Figure B 3/4.4-1

Fast Neutron Fluence (E > 1 Mev) as a Function of Full Power Service Life

0 0341

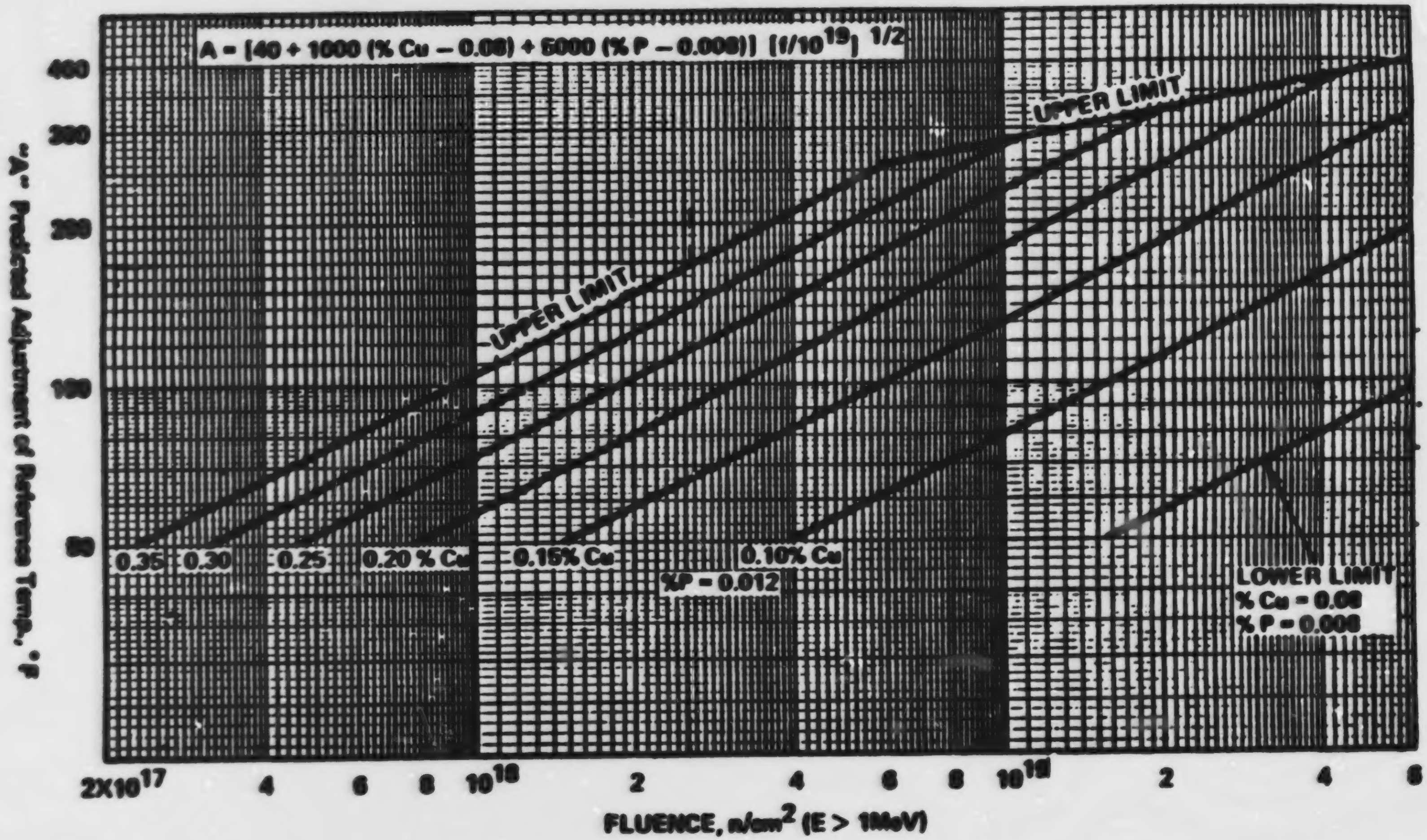

Figure B 3/4.4-2 Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content
For Copper and Phosphorus Contents Other Than Those Plotted, Use the Expression for "A" Given on the Figure

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

Given the copper and phosphorous contents of the most limiting material, the radiation-induced ΔRT_{NDT} can be estimated from Figure B 3/4.4-2. Fast-neutron fluence ($E > 1$ Mev) at the 1/4 T (wall thickness) and 3/4 T (wall thickness) vessel locations are given as a function of full-power service life in Figure B 3/4.4-1. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to insure that no other component will be limiting with respect to RT_{NDT}.

The preirradiation fracture-toughness properties of the Beaver Valley Unit 2 reactor vessel materials are presented in Table B 3/4.4-1. The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the 1972 Summer Addenda to Section III of the ASME Boiler and Vessel Code.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup and cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code.[2] The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T-RT_{NDT} + 160)] \qquad (4\text{-}1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal reference nilductility temperature, RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G to the ASME Code[2] as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (4\text{-}2)$$

where

K_{IM} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is a function of temperature to the RT_{NDT} of the material

$C = 2.0$ for Level A and Level B service limits

$C = 1.5$ for hydrostatic and leak test conditions during which the reactor core is not critical

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[2] ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendices, "Rules for Construction of Nuclear Vessels," Appendix G. "Protection Against Nonductile Failure," pp. 559-569, 1980 Edition, American Society of Mechanical Engineers, New York, 1983.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factors, K_{It}, for the reference flaw are computed. From equation 4-2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure-versus-coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increases with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit ensures the constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and insures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack during heatup is

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same coolant temperature.

During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temeprature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion. Then, composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and auxiliary spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 50°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

OVERPRESSURE PROTECTION SYSTEMS

The Maximum Allowed PORV Setpoint for the Overpressure Protection Systems (OPPS) is derived by analysis which models the performance of the OPPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that nominal 10 EFPY Appendix G limits will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for OPPS; (3) instrument uncertainties; and (4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE is allowed for up to 4 hours. As shown by analysis LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single LHSI pump with no credit for accumulator injection. Given the short time duration

OVERPRESSURE PROTECTION SYSTEMS (Continued)

that the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one LHSI pump). For temperatures above 325°F, an overpressure event occurring as a result of starting these two pumps can be successfully mitigated by operation of both PORVs without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

The maximum allowed PORV setpoint for the Overpressure Protection System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g)(6)(i).

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3/4.4.11 REACTOR COOLANT SYSTEM RELIEF VALVES

The relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path. The operability of at least one PORV will ensure the additional capability to vent the pressurizer steam space via the PORV's.

3/4.4.12 REACTOR COOLANT SYSTEM HEAD VENTS

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head or the pressurizer steam space via the PORV's ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System Head vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each of the RCS accumulators ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the accident analysis are met.

The limit of one hour for operation with an inoperable accumulator minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures.

The RCS accumulators are isolated when RCS pressure is reduced to 1000 ± 100 psig to prevent borated water from being injected into the RCS during normal plant cooldown and depressurization conditions and also to prevent inadvertent overpressurization of the RCS at reduced RCS temperature.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

The surveillance requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained.

The limitation for a maximum of one charging pump to be OPERABLE and the surveillance requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a . As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq$ 0.75 La during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 INTERNAL PRESSURE AND AIR TEMPERATURE

The limitations on containment internal pressure and average air temperature as a function of RWST temperature ensure that 1) the containment structure is prevented from exceeding its design negative pressure of 8.0 psia, 2) the containment peak pressure does not exceed the design pressure of 45 psig during LOCA conditions, and 3) the containment pressure is returned to subatmospheric conditions following a LOCA.

The containment internal pressure and temperature limits shown as a function of RWST and service water temperature describe the operational envelope that will 1) limit the containment peak pressure to less than its design value

3/4.6.1.4 AND 3/4.6.1.5 INTERNAL PRESSURE AND AIR TEMPERATURE (Continued)

of 45 psig and 2) ensure the containment internal pressure returns subatmospheric within 60 minutes following a LOCA.

The limits on the parameters of Figure 3.6-1 are consistent with the assumptions of the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 44.7 psig in the event of a LOCA. The visual and Type A leakage tests are sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH AND RECIRCULATION SPRAY SYSTEMS

The OPERABILITY of the containment spray systems ensures that containment depressurization and subsequent return to subatmospheric pressure will occur in the event of a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the accident analyses.

3/4.6.2.3 CHEMICAL ADDITION SYSTEM

The OPERABILITY of the chemical addition system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for both a LOCA and major secondary system breaks.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

3/4.6.5.1 STEAM JET AIR EJECTOR

The closure of the manual isolation valves in the suction of the steam jet air ejector ensures that 1) the containment internal pressure may be maintained within its operation limits by the mechanical vacuum pumps and 2) the containment atmosphere is isolated from the outside environment in the event of a LOCA. These valves are required to be closed for containment isolation.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition and Winter 1972 Addenda. The total relieving capacity for all valves on all of the steam lines is 12.7 x 10^6 lbs/hr which is 110 percent of the total secondary steam flow of 11.6 x 10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip reductions are derived on the following bases:

For N loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

(109) = Power Range Neutron Flux-High Trip Setpoint for (N) loop operation

3/4.7.1.1 SAFETY VALVES (Continued)

X = Total relieving capacity of all safety valves per steam line in lbs/hour (4,242,375)

Y = Maximum relieving capacity of one safety valve in lbs/hour (848,475)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 350 gpm at a pressure of 1133 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1133 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 PRIMARY PLANT DEMINERALIZED WATER (PPDW)

The OPERABILITY of the PPDW storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 0.35 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator average impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the primary component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects or accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related.equipment without

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3/4.7.5 ULTIMATE HEAT SINK (Continued)

exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants."

3/4.7.6 FLOOD PROTECTION

The limitation on flood level ensures that facility operation will be terminated in the event of flood conditions. The limit of elevation 695 Mean Sea Level was selected on an arbitrary basis as an appropriate flood level at which to terminate further operation and initiate flood protection measures for safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

The OPERABILITY of the control room emergency air cleanup and pressurization system ensures that the control room will remain habitable with respect to potential radiation and chlorine hazards for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The control room air cleanup system includes two pressurization systems. The filtration pressurization system draws outside air through filters. The bottled air pressurization system pressurizes by discharge of air from bottles without filtration and with closure of intake and exhaust dampers. Although the bottles are shared with Unit 1, the discharge can be initiated by Unit 2 control systems in response to chlorine or radiation levels. Closure of the intake and exhaust dampers can be initiated by Unit 2 control systems. However, closure of dampers in one intake and in one exhaust is dependent upon availability of Unit 1 power sources.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting from a Fuel Handling Accident (FHA) and from leakage of loss of coolant accident (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building penetrations into the contiguous areas ventilated by the SLCRS except for the Emergency Air Lock. The operation of this system was assumed in calculating the postulated offsite doses in the analysis for a FHA. System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, no credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage even though an unquantifiable amount of contiguous area penetration leakage would in fact be collected and filtered. Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA and leakage resulting from a FHA will not exceed 10 CFR 100 limits.

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3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

3/4.7.10 and 3/4.7.11 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

Deleted

3/4.7.12 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other similar event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified OPERABLE by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically, susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

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SNUBBERS (Continued)

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling or 18 month intervals not to exceed two (2) years. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber life review are not intended to affect plant operation.

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SWE)

The OPERABILITY of the SWE ensures that sufficient cooling capacity is available to bring the reactor to a cold shutdown condition in the event that a barge explosion at the station's intake structure or any other extremely remote event would render all of the normal Service Water System supply pumps inoperable.

3/4.8.1, 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based on the recommendations of Regulatory Guides 1.9, Revision 2, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979, Appendix A to Generic Letter 84-15 and Generic Letter 83-26, "Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests."

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

3/4.8.1, 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION (Continued)

Table 3.8-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 3.8-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; 3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and 4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

Isolating all reactor water makeup paths from unborated water sources precludes the possibility of an uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This limitation is consistent with the initial conditions assumed in the accident analyses for MODE 6.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure limit leakage of radioactive material within containment to the environment to ensure compliance with 10 CFR 100 limits. The requirements on operation of the SLCRS ensure that trace amounts of radioactive material within containment will be filtered through HEPA filters charcoal absorbers prior to discharge to the atmosphere. These requirements are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirements for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

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3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies; 2) each crane has sufficient load capacity to lift a control rod or fuel assembly; and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the normal weight of a fuel assembly over other fuel assemblies ensures that no more than the contents of one fuel assembly plus an additional 50 rods in the struck fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained throughout the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The integrity of the containment penetrations of this system is required to meet 10 CFR 100 requirements in the event of a fuel handling accident inside containment. Applicability in MODE 5, although not an NRC safety requirement, will provide additional protection against small releases of radioactive material from the containment during maintenance activities.

BASES

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 and 3/4.9.13 FUEL BUILDING VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses. The spent fuel pool area ventilation system is non-safety related and only recirculates air through the fuel building. The fuel building portion of the SLCRS is safety related and continuously filters the fuel building exhaust air. This maintains a negative pressure in the fuel building.

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 and the RCS T_{avg} may fall slightly below the minimum temperature of Specification 3.1.1.5.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception is required to perform certain startup tests.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Position Indication System to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposure within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to an individual and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October, 1977, and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

This specification applies to the release of liquid effluents from Beaver Valley Power Station, Unit No. 2. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID WASTE TREATMENT

The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents. This specification applies to Beaver Valley Power Station, Unit No. 2.

3/4.11.1.4 LIQUID HOLDUP TANKS

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at anytime at the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B. Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to $\leq$ 500 mrem/year to the total body or to $\leq$ 3,000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to $\leq$ 1,500 mrem/year.

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3/4.11.2.1 DOSE RATE (Continued)

This specification applies to the release of gaseous effluents from Beaver Valley Power Station, Unit No. 2. For units with shared radwaste treatment system, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.2 DOSE, NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the release of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October, 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July, 1977. The ODCM equations provided for determining the air doses at the exclusion area boundary are based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111. This specifications applies to the release of gaseous effluents from Beaver Valley Power Station, Unit No. 2.

3/4.11.2.3 DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I.

The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A. of Appendix I that conformance with the guides of Appendix I be

3/4.11.2.3 DOSE, RADIOIODINES RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES (Continued)

shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routing Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October, 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July, 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form, and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man. This specification applies to radioactive material in particulate form and radionuclides other than noble gases released from Beaver Valley Power Station, Unit No. 2.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents. This specification applies to gaseous radwaste from Beaver Valley Power Station, Unit No. 2

3/4.11.2.5 GASEOUS WASTE STORAGE TANKS

Restricting the quantity of radioactivity contained in any connected group of gaseous waste storage tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting total body exposure to an individual located at the nearest exclusion area boundary for two hours immediately following the onset of the release will not exceed 0.5 rem. The

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3/4.11.2.5 GASEOUS WASTE STORAGE TANKS (Continued)

specified limit restricting the quantity of radioactivity contained in any connected group of gaseous waste storage tanks was specified to ensure that the total body exposure resulting from the postulated release remained a suitable fraction of the reference value setforth in 10 CFR 100.11(a)(1). The curie content limit is applied individually to each gaseous waste storage tank and collectively to the number of unisolated gaseous waste storage tanks.

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Isolation of the affected tank for purposes of purging and/or discharge permits the flammable gas concentrations of the tank to be reduced below the lower explosive limit in a hydrogen rich system. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A of 10 CFR Part 50 and requires the system be used whenever solid radwastes require processing and packaging prior to being shipped offsite. The process parameters used in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The Specification requires the preparation and submittal of a Special Report, in lieu of any other report, whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 nuclear reactors, it is highly unlikely that the resultant dose to MEMBER(S) OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to MEMBER(S) OF THE PUBLIC for the calendar year to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to MEMBER(S) OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles be considered.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of MEMBER(S) OF THE PUBLIC resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirements of 40 CFR 141.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring programs are made if required by the results of this census. The best survey information from the door-to-door survey, aerial survey or by consulting with local agriculture authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used: 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an Interlaboratory Comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.1 The site boundary for gaseous effluents shall be as shown in Figure 5.1-1. Release paths are shown on Figure 5.1-2.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.2 The site boundary for liquid effluents shall be as shown in Figure 5.1-1. Release points are shown on Figure 5.1-2.

EXCLUSION AREA

5.1.3 The exclusion area shall be as shown in Figure 5.1-3.

LOW POPULATION ZONE

5.1.4 The low population zone shall be as shown in Figure 5.1-4.

FLOOD CONTROL

5.1.5 The flood control provisions (dikes, levees, etc.) shall be designed and maintained in accordance with the original design provisions contained in Section 3.4.1 of the FSAR.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 126 feet.

b. Nominal inside height = 185 feet.

c. Minimum thickness of concrete walls = 4.5 feet.

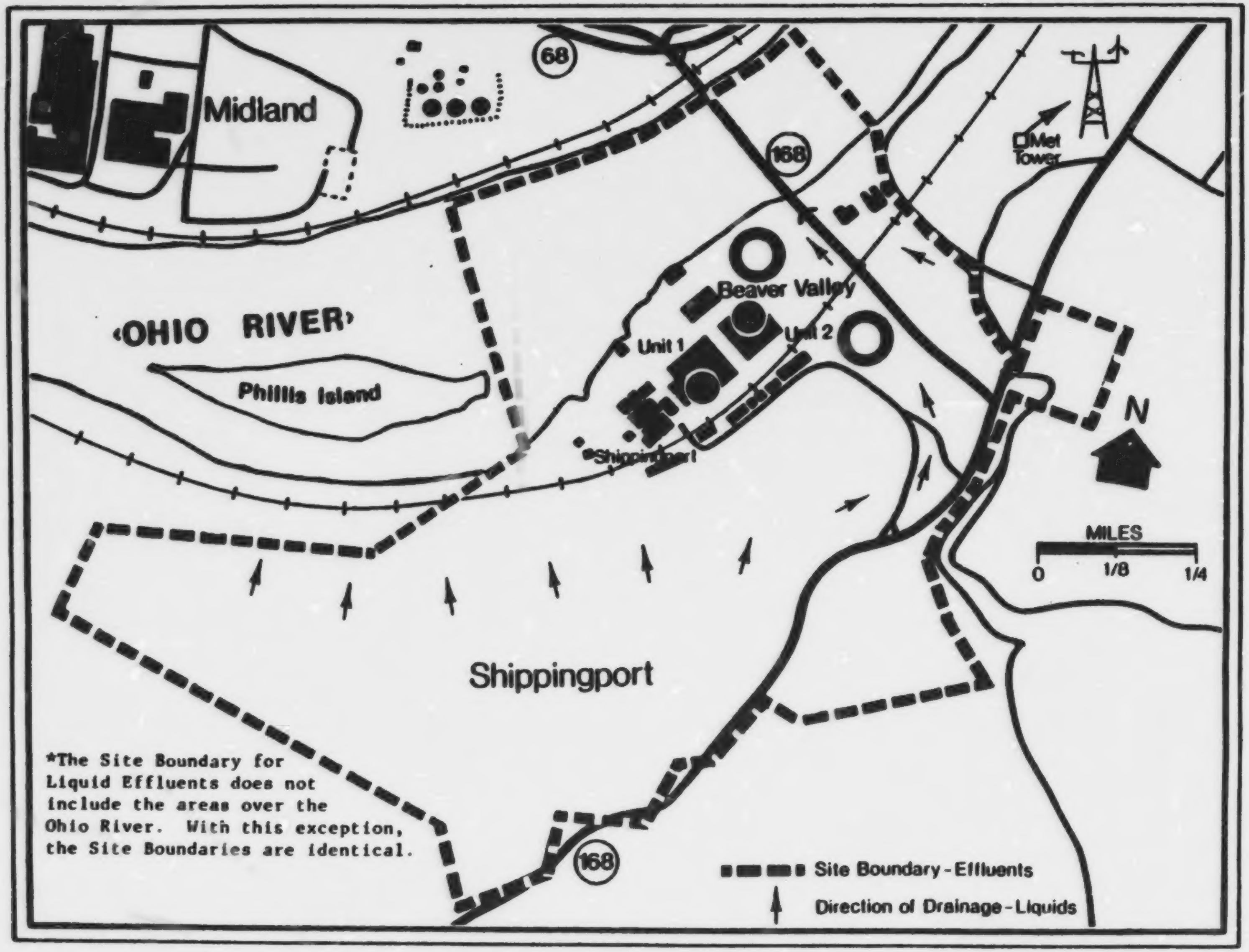
d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of foundation mat = 10 feet.

f. Nominal thickness of vertical portion of steel liner = 3/8 inch.

g. Nominal thickness of steel liner, dome portion = 1/2 inch.

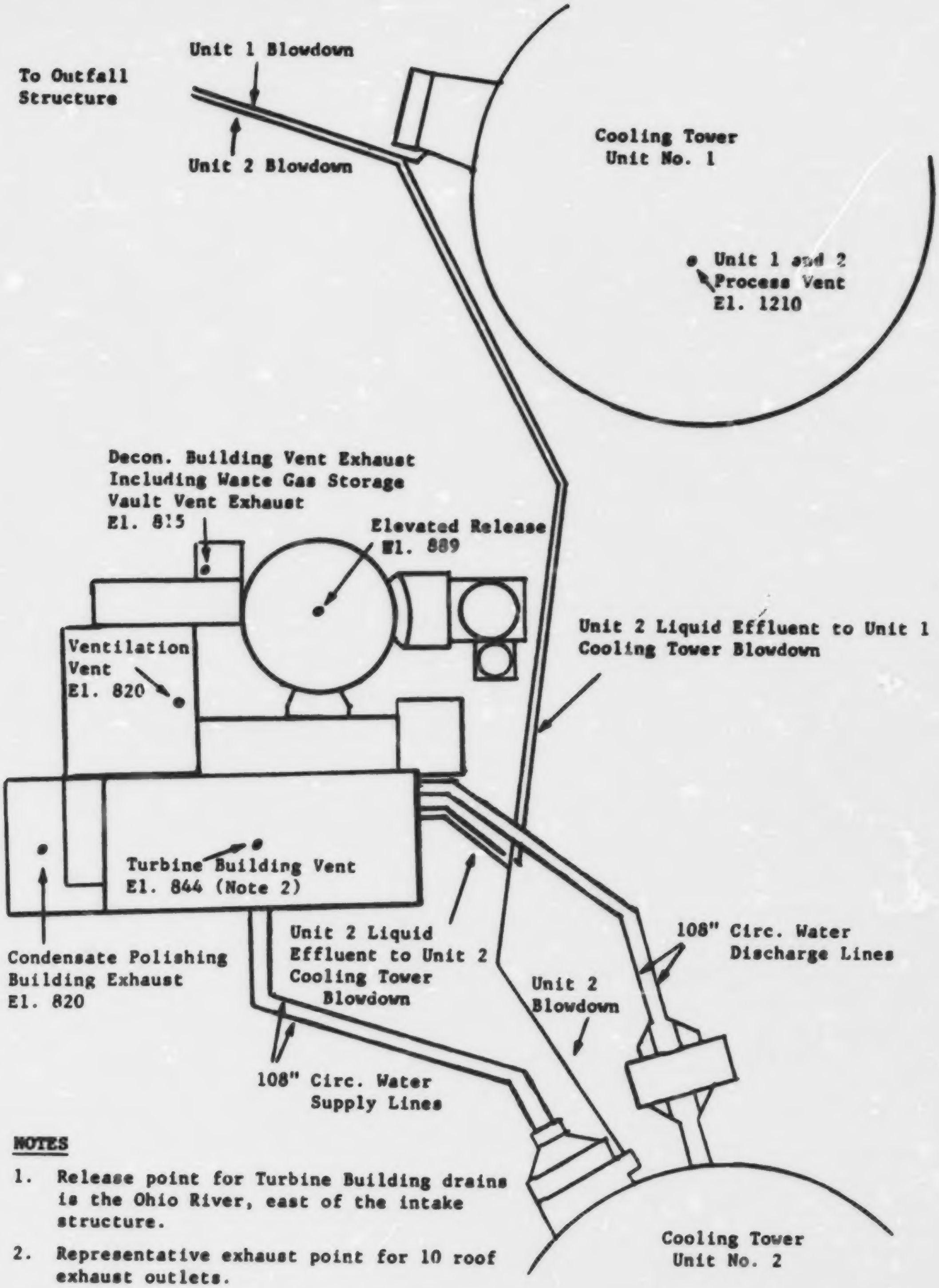
h. Minimum free volume = 1.73 x 10^6 cubic feet.



*The Site Boundary for Liquid Effluents does not include the areas over the Ohio River. With this exception, the Site Boundaries are identical.

FIGURE 5.1-1

SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS
FOR THE BEAVER VALLEY POWER STATION UNIT 2


To Outfall Structure
Unit 1 Blowdown
Unit 2 Blowdown
Cooling Tower Unit No. 1
Unit 1 and 2 Process Vent El. 1210
Decon. Building Vent Exhaust Including Waste Gas Storage Vault Vent Exhaust El. 815
Elevated Release El. 889
Ventilation Vent El. 820
Unit 2 Liquid Effluent to Unit 1 Cooling Tower Blowdown
Turbine Building Vent El. 844 (Note 2)
Condensate Polishing Building Exhaust El. 820
Unit 2 Liquid Effluent to Unit 2 Cooling Tower Blowdown
108" Circ. Water Discharge Lines
Unit 2 Blowdown
108" Circ. Water Supply Lines
Cooling Tower Unit No. 2

NOTES

1. Release point for Turbine Building drains is the Ohio River, east of the intake structure.
2. Representative exhaust point for 10 roof exhaust outlets.

FIGURE 5.1-2

RADIOACTIVE GASEOUS AND LIQUID RELEASE POINTS - BEAVER VALLEY POWER STATION UNIT 2

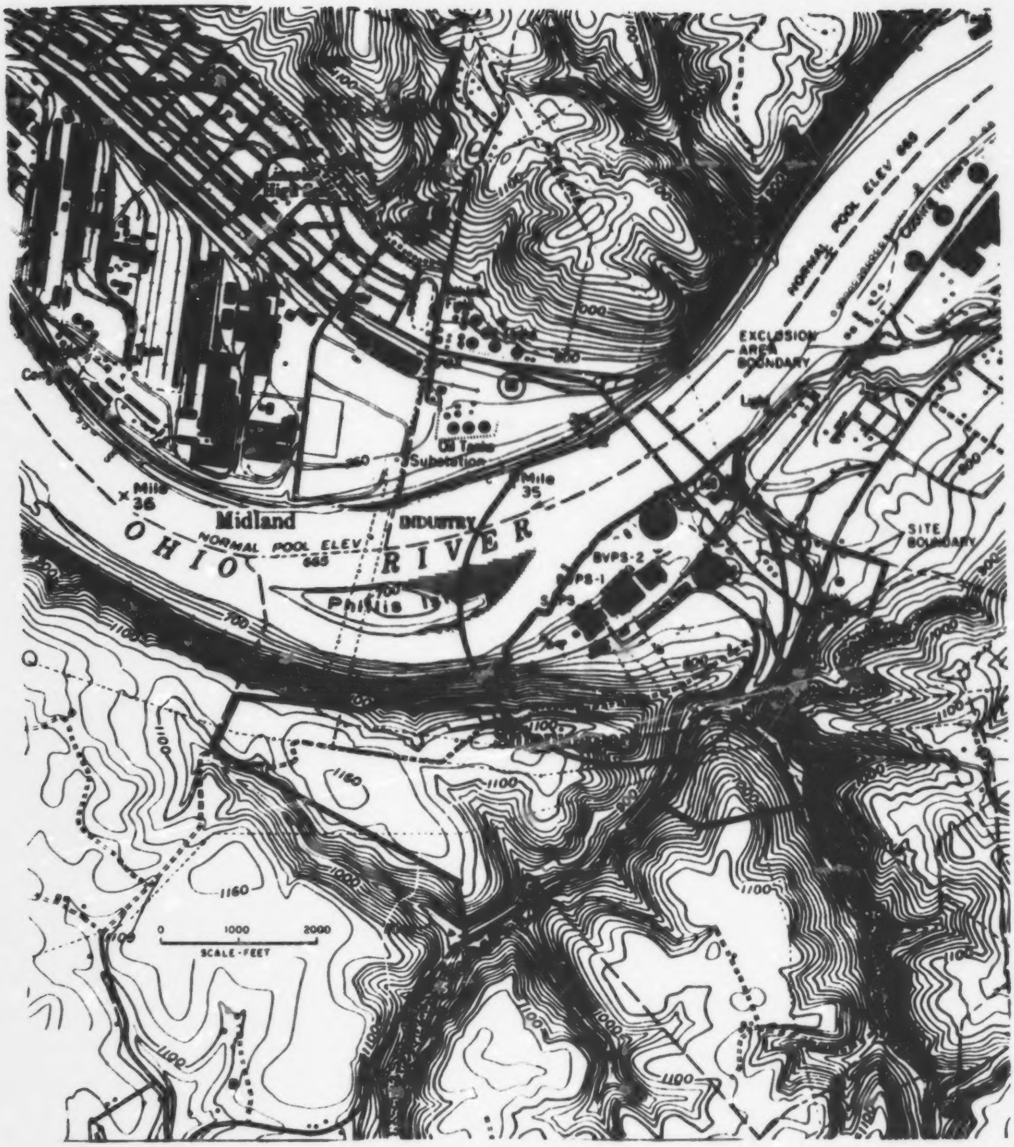


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FIGURE 5.1-3

EXCLUSION AREA - BEAVER VALLEY POWER STATION UNIT 2

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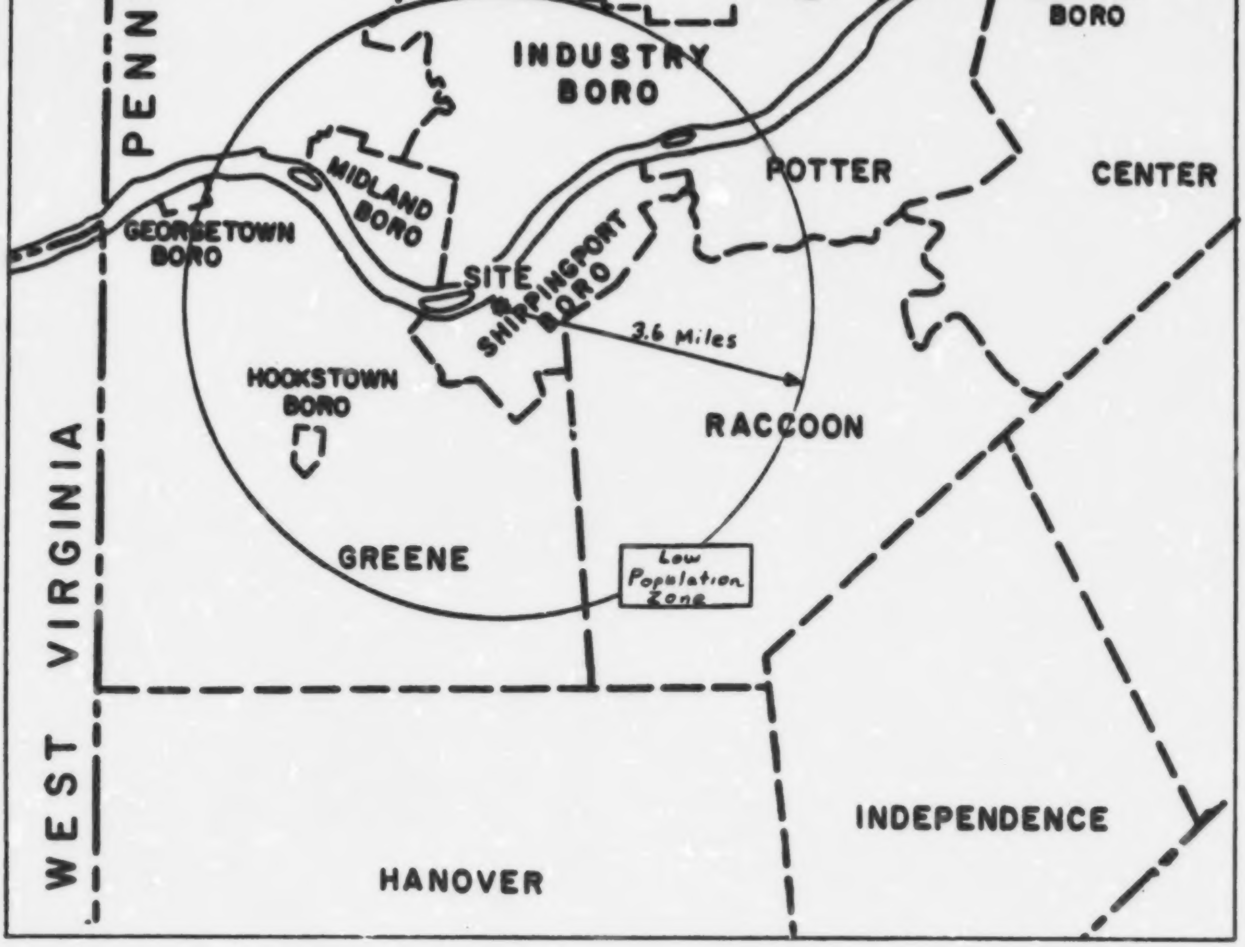

FIGURE 5.1-4

LOW POPULATION ZONE - BEAVER VALLEY POWER STATION UNIT 2

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DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for maximum internal pressure of 45 psig and a temperature of 280.0°F.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.3 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9370 cubic feet at a nominal T_{avg} of 576°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with a minimum of 10.4375 inch center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.95 with the storage pool filled with unborated water. The k_{eff} of ≤ 0.95 includes a conservative allowance of at least 1.4% $\Delta k/k$ for uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 751'-3".

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1088 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I items in Section 3.7 of the FSAR shall be designed and maintained to the original design provisions with allowance for normal degradation pursuant to the applicant Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower shall be located as shown on Figure 5.1-1.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The corporate organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; senior reactor operators, reactor operators, radiation control technicians, auxiliary operators, meter and control repairman, and all personnel actually performing work on safety related equipment.

The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

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FACILITY STAFF (Continued)

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or predesignated alternate, or higher levels of management. Authorized deviations to the working hour guidelines shall be documented and available for NRC review.

6.2.3 INDEPENDENT SAFETY EVALUATION GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to corporate management. If not otherwise implemented, all recommendations shall then be made to the Vice President, Nuclear Group.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and and forwarded each calendar month to the Vice President, Nuclear Group.

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*Not responsible for sign-off function.

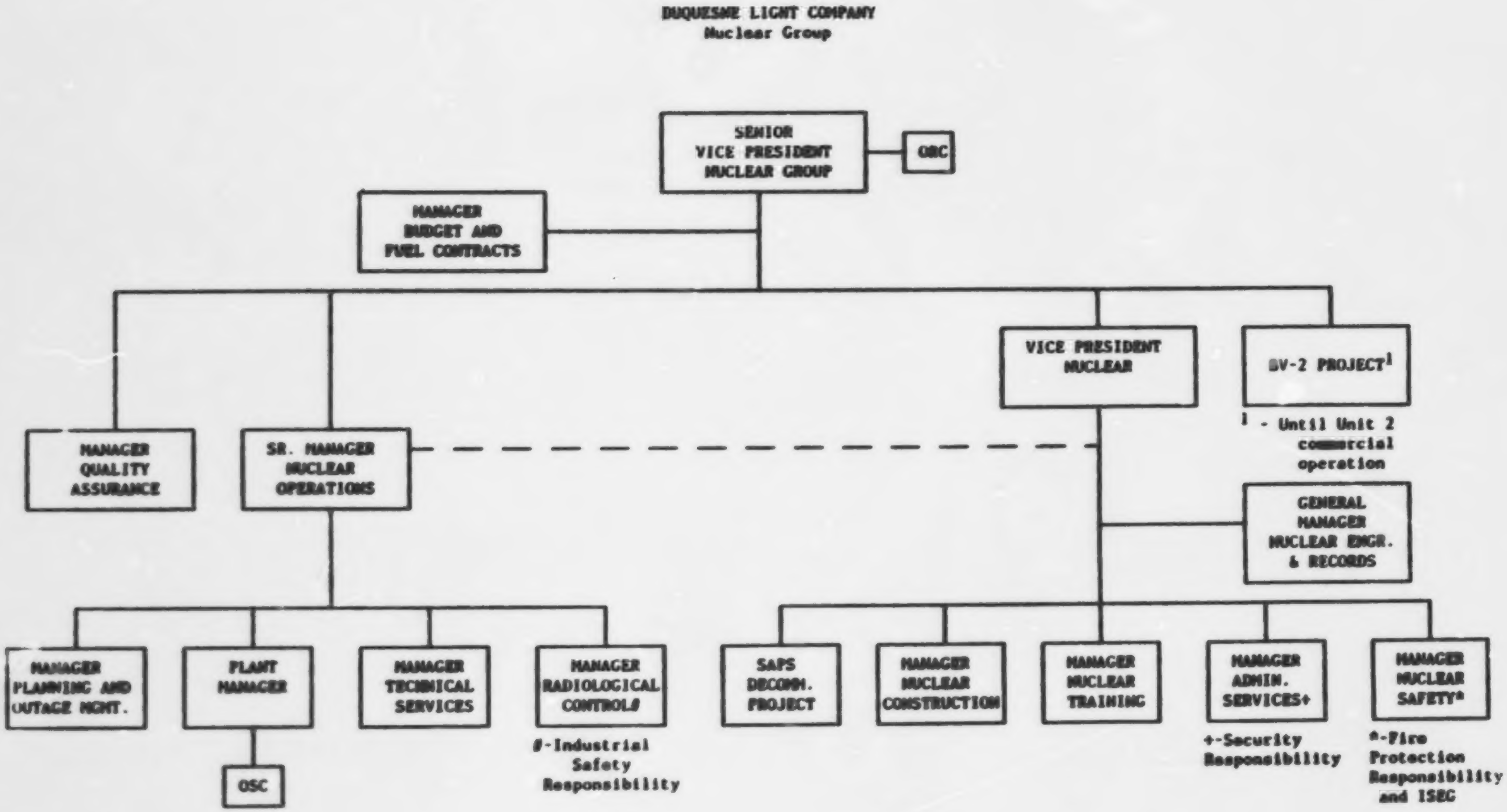
DUQUESNE LIGHT COMPANY
Nuclear Group
SENIOR VICE PRESIDENT NUCLEAR GROUP
ORC
MANAGER BUDGET AND FUEL CONTRACTS
VICE PRESIDENT NUCLEAR
BV-2 PROJECT[1]
1 - Until Unit 2 commercial operation
MANAGER QUALITY ASSURANCE
SR. MANAGER NUCLEAR OPERATIONS
GENERAL MANAGER NUCLEAR ENGR. & RECORDS
MANAGER PLANNING AND OUTAGE MGMT.
PLANT MANAGER
OSC
MANAGER TECHNICAL SERVICES
MANAGER RADIOLOGICAL CONTROL#
#-Industrial Safety Responsibility
SAPS DECOMM. PROJECT
MANAGER NUCLEAR CONSTRUCTION
MANAGER NUCLEAR TRAINING
MANAGER ADMIN. SERVICES+
+-Security Responsibility
MANAGER NUCLEAR SAFETY*
*-Fire Protection Responsibility and ISEG

FIGURE 6.2-1

CORPORATE ORGANIZATION (PARTIAL)

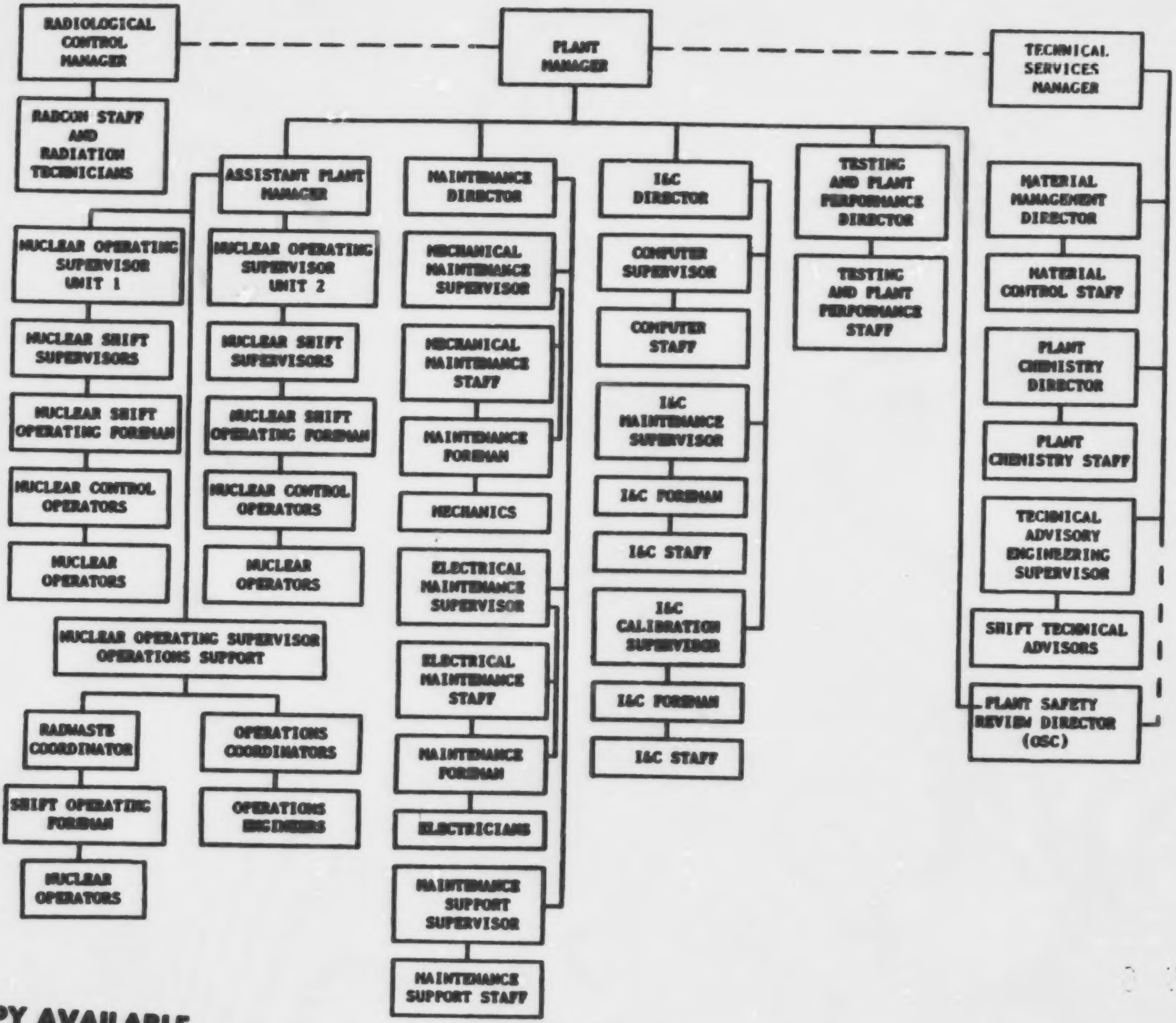
RADIOLOGICAL CONTROL MANAGER
PLANT MANAGER
TECHNICAL SERVICES MANAGER
RADCON STAFF AND RADIATION TECHNICIANS
ASSISTANT PLANT MANAGER
MAINTENANCE DIRECTOR
I&C DIRECTOR
TESTING AND PLANT PERFORMANCE DIRECTOR
MATERIAL MANAGEMENT DIRECTOR
NUCLEAR OPERATING SUPERVISOR UNIT 1
NUCLEAR OPERATING SUPERVISOR UNIT 2
MECHANICAL MAINTENANCE SUPERVISOR
COMPUTER SUPERVISOR
TESTING AND PLANT PERFORMANCE STAFF
MATERIAL CONTROL STAFF
NUCLEAR SHIFT SUPERVISORS
NUCLEAR SHIFT SUPERVISORS
MECHANICAL MAINTENANCE STAFF
COMPUTER STAFF
PLANT CHEMISTRY DIRECTOR
NUCLEAR SHIFT OPERATING FOREMAN
NUCLEAR SHIFT OPERATING FOREMAN
MAINTENANCE FOREMAN
I&C MAINTENANCE SUPERVISOR
PLANT CHEMISTRY STAFF
NUCLEAR CONTROL OPERATORS
NUCLEAR CONTROL OPERATORS
MECHANICS
I&C FOREMAN
TECHNICAL ADVISORY ENGINEERING SUPERVISOR
NUCLEAR OPERATORS
NUCLEAR OPERATORS
ELECTRICAL MAINTENANCE SUPERVISOR
I&C STAFF
NUCLEAR OPERATING SUPERVISOR OPERATIONS SUPPORT
ELECTRICAL MAINTENANCE STAFF
I&C CALIBRATION SUPERVISOR
SHIFT TECHNICAL ADVISORS
I&C FOREMAN
PLANT SAFETY REVIEW DIRECTOR (OSC)
RADWASTE COORDINATOR
OPERATIONS COORDINATORS
MAINTENANCE FOREMAN
I&C STAFF
SHIFT OPERATING FOREMAN
OPERATIONS ENGINEERS
ELECTRICIANS
NUCLEAR OPERATORS
MAINTENANCE SUPPORT SUPERVISOR
MAINTENANCE SUPPORT STAFF

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FIGURE 6.2-2

FACILITY ORGANIZATION

TABLE 6.2.1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

LICENSE CATEGORY QUALIFICATIONS	APPLICABLE MODES	
	1, 2, 3 and 4	5 and 6
SRO*	2	1**
RO	2	1
Non-Licensed Auxiliary Operator	2	1
Shift Technical Advisor	1(a)	None Required

*Includes the Licensed Senior Reactor Operator serving as the Shift Supervisor.

**Does not include the Licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE OPERATIONS.

#Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(a)The Shift Technical Advisor (STA) position may be filled by the same STA from the BVPS Unit 1, if the individual is qualified for BVPS Unit 2.

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6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility and Radiation Protection staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiological Control Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the technical advisory engineering representative who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 ONSITE SAFETY COMMITTEE (OSC)

FUNCTION

6.5.1.1 The OSC shall function to advise the Plant Manager on all matters related to nuclear safety and shall provide review capability in the areas of:

a. nuclear power plant operations
b. radiological safety
c. maintenance
d. nuclear engineering
e. nuclear power plant testing
f. technical advisory engineering
g. chemistry
h. quality control
i. instrumentation and control

COMPOSITION

6.5.1.2 The plant Safety Review Director is the OSC Chairman and shall appoint all members of the OSC. The membership shall consist of a minimum of one individual from each of the ares designated in 6.5.1.1.

OSC members and alternates shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The nuclear power plant operations individual shall meet the qualifications of Section 4.2.2 and the maintenance individual shall meet the qualifications of Section 4.2.3.

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COMPOSITION (Continued)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The OSC shall meet at least once per calendar month and as convened by the OSC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the OSC shall consist of the Chairman or his designated alternate and at least one half of the members including alternates.

RESPONSIBILITIES

6.5.1.6 The OSC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes of intent thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Senior Manager Nuclear Operations and to the Chairman of the Offsite Review Committee.

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Offsite Review Committee.

AUTHORITY

6.5.1.7 The OSC Shall:

a. Recommend to the Plant Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Senior Manager Nuclear Operations and the Offsite Review Committee of disagreement between the OSC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1. above.

RECORDS

6.5.1.8 The OSC shall maintain written minutes of each meeting and copies shall be provided to the Senior Manager Nuclear Operations and Chairman of the Offsite Review Committee.

6.5.2 OFFSITE REVIEW COMMITTEE (ORC)

FUNCTION

6.5.2.1 The ORC shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The chairman and all members of the ORC shall be appointed by the Senior Vice President, Nuclear Group. The membership shall consist of a minimum of five individuals who collectively possess a broad based level of experience and competence enabling the committee to review and audit those activities designated in 6.5.2.1 above and to recognize when it is necessary to obtain technical advice and counsel. An individual may possess expertise in more than one speciality area. The collective competence of the committee will be maintained as changes to the membership are made.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the ORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ORC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the ORC Chairman to provide expert advice to the ORC.

MEETING FREQUENCY

6.5.2.5 The ORC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of ORC shall consist of the Chairman or his designated alternate and at least four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The ORC shall review:

a. The safety evaluations for 1) changes to procedures, equipment, or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not consitute an unreviewed safety question.

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REVIEW (Continued)

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

i. Reports and meeting minutes of the OSC.

j. The results of the Radiological Monitoring Program prior to submittal of the annual report provided in accordance with Specification 6.9.1.10.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the ORC. These audits shall encompass:

a. The conformance of facility operations to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months.

d. The performance activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 12 months.

AUDITS (Continued)

f. The Facility Security Plan and implementing procedures at least once per 12 months.

g. Any other area of facility operation considered appropriate by the ORC or the Vice President, Nuclear.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

AUTHORITY

6.5.2.9 The ORC shall report to and advise the Senior Vice President, Nuclear Group on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of ORC activities shall be prepared, approved, and distributed as indicated by the following:

a. Minutes of each ORC meeting shall be prepared for and approved by the ORC Chairman or Vice Chairman within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be documented in the ORC meeting minutes.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President, Nuclear Group and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified in accordance with 10 CFR 50.72 and/or a report be submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the OSC, and the results of this review shall be submitted to the ORC.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one (1) hour.

b. The Safety Limit violation shall be reported to the Commission within one hour. The Safety Limit violation shall be reported to the Senior Manager Nuclear Operations and to the ORC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the On-Site Safety Committee (OSC). This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the ORC and the Senior Manager Nuclear Operations within 30 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above and changes thereto, shall be reviewed by the OSC and approved by the Plant Manager, predesignated alternate or a predesignated Manager to whom the Plant Manager has assigned in writing the responsibility for review and approval of specific subjects considered by the committee, as applicable. Changes to procedures and administrative policies of 6.8.1 above that do not receive OSC review, such as correcting typographical errors, reformatting procedures and other changes not affecting the purpose for which the procedure is performed shall receive an independent review by a qualified individual and approved by a designated manager or director.

PROCEDURE (Continued)

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the OSC and approved by the Plant Manager within 14 days of implementation.

6.8.4 A Post-Accident monitoring program shall be established, implemented, and maintained. The program will provide the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples following an accident. The program shall include the following:

(i) Training of personnel,
(ii) Procedures for sampling and analysis, and
(iii) Provisions for maintenance of sampling and analysis equipment.

6.8.5 A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation shall be implemented. This program shall be described in the station chemistry manual and shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to measure the values of the critical parameters;

c. Identification for process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for off control point chemistry conditions; and

f. A procedure identifying:

1) the authority responsible for the interpretation of the data, and

2) the sequence and timing of administrative events required to initiate corrective action.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing will be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or had been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operations, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation of the number of station, utility, and othe personnel (including contractors) receiving exposure greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions[2] (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

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[1] A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site.

[2] This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

ANNUAL REPORTS (Continued)

b. Documentation of all challenges to the pressurizer power operated relief valves (PORVS) or pressurizer safety valves.

c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded: (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC no later than the 15th of each month following the calendar month covered by the report.

6.9.1.7 This item intentionally blank

6.9.1.8 This item intentionally blank

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ANNUAL RADIOLOGICAL ENVIRONMENTAL REPORT[3]

6.9.1.10 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year and will include reporting any deviations not reported under 6.9.2 with respect to the Radiological Effluent Technical Specifications.

6.9.1.11 The annual radiological environmental reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the

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[3] A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to both units.

TABLE 6.9.-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility____________________ Docket No.__________

Location of Facility____________________ Reporting Period_________

(County,State)

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limits of Detection[a] (LLD)	All Indicator Locations Mean (1)[b] Range[b]	Locations with Highest Annual Mean		Control Locations Mean (1)[b] Range[b]	Number of REPORTABLE OCCURRENCES
				Name Distance and Direction	Mean (1)[b] Range[b]		

[a] Nominal Lower Limits of Detection (LLD) as defined in table notation a. of Table 4.11-1 of Specification 4.11.1

[b] Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in Parentheses (1).

ANNUAL RADIOLOGICAL ENVIRONMENTAL REPORT (Continued)

monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Table 6.9-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: A summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program required by Specification 3.12.3.

SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[4]

6.9.1.12 Routine radioactive effluent release reports covering the operating of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

6.9.1.13 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," with data summarized on a quarterly basis following the format of Appendix B thereof.

In addition the radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This report shall also include an assessment of the radiation doses from radioactive effluents to MEMBER(S) OF THE PUBLIC due to their activities inside the site boundary (Figure 5.1-1 and 5.1-2) during the report period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with OFFSITE DOSE CALCULATION MANUAL (ODCM).

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[4] A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed real individual from reactor releases for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1. The SKYSHINE Code (available from Radiation Shielding Information Center, (ORNL) is acceptable for calculating the dose contribution from direct radiation due to N-16.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter as outlined in Regulatory Guide 1.21. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with ODCM.

The radioactive effluent release reports shall also include any licensee initiated changes to the ODCM made during the 6 month period.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.14 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the Regional Office of the NRC, with a copy to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to suport F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

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SPECIAL REPORTS (Continued)

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Sealed source leakage in excess of limits, Specification 4.7.9.1.3.

f. Miscellaneous reporting requirements specified in the Action Statements for Radiological Effluent Technical Specifications.

g. Containment Inspection Report, Specification 4.6.1.6.2.

h. Steam generator tube inservice inspection, Specification 4.4.5.5.

i. Inoperable accident monitoring, Specification 3.3.3.8.

j. Operation of PORVs or RCS vents, Specification 3.4.9.3.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five (5) years;

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

RECORD RETENTION (Continued)

b. Records of new irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSC and the ORC.

l. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the Radiological Environmental Monitoring Program.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiological

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HIGH RADIATION AREA (Continued)

Work Permit* or Radiological Access Control Permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by a facility health physics supervisor in the Radiological Work Permit or Radiological Access Control Permit.

6.12.2 The requirements of 6.12.1, above, also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or a facility health physics supervisor.

6.13 This item intentionally blank

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*Health physics personnel, or personnel escorted by health physics personnel in accordance with approved emergency procedures, shall be exempt from the RWP issuance requirement during the performance of their radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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6.14 PROCESS CONTROL PROGRAM (PCP)

FUNCTION

6.14.1 The PCP shall be those manuals, procedures, or references to procedures containing the processing steps, a set of established process parameters and the steps detailing the program of sampling, analysis, and evaluation within which solidification of radioactive wastes is assured, consistent with Specification 3.11.3.1 and the surveillance requirements of these Technical Specifications.

6.14.2 Licensee initiated changes

1. Shall become effective upon review and acceptance by the OSC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

FUNCTION

6.15.1 The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

6.15.2 Licensee initiated changes:

1. Shall become effective upon review and acceptance by the OSC.

6.16 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

FUNCTION

6.16.1 The radioactive waste treatment systems (liquid, gaseous and solid are those systems described in the facility Final Safety Analysis Report or Hazards Summary Report, and amendments thereto, which are used to maintain that control over radioactive materials in gaseous and liquid effluents and in solid waste packaged for offsite shipment required to meet the LCO's set forth in Specifications 3.11.1.1, 3.11.1.2, 3.11.1.3, 3.11.1.4, 3.11.2.1, 3.11.2.2, 3.11.2.3, 3.11.2.4, 3.11.2.5, 3.11.2.6, 3.11.3.1 and 3.11.4.1.

6.16.2 MAJOR CHANGES as defined in Section 1 to the radioactive waste systems (liquid, gaseous and solid) shall be made by the following method:

A. Licensee initiated changes:

   1. If a permanent facility change is made to a radioactive treatment system that could result in an increase in the volume or activity discharged, the Commission shall be informed by the inclusion of a

FUNCTION (Continued)

suitable discussion of each change in the Annual 10 CFR 50.59 Report for the period in which the changes were made. The discussion of each change shall contain:

a. A summary of the evaluation that led to the determination that the change could be made (in accordance with 10 CFR 50.59);

b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change will be submitted which shows the predicted increase of releases of radioactive materials in liquid or gaseous effluents and/or quantity of solid waste from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change which shows the expected increase in the maximum exposures to an individual in the unrestricted area from those previously predicted in the license application and amendments thereto;

f. A comparison of the predicted increase of releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period the changes were made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the OSC.

2. The change shall become effective upon review and acceptance by the OSC.

6.16.3 Background of what constitutes MAJOR CHANGES to radioactive waste systems (liquid, gaseous, and solid).

A. Background

1. 10 CFR Part 50, Section 50.34a(a) requires that each application to construct a nuclear power reactor provide a description of the equipment installed to maintain control over radioactive material in gaseous and liquid effluents produced during normal reactor operations including operational occurrences.

FUNCTION (Continued)

2. 10 CFR Part 50, Section 50.34a(b)(2) requires that each application to construct a nuclear power reactor provide an estimate of the quantity of radionuclides expected to be released annually to unrestricted areas in liquid and gaseous effluents produced during normal reactor operation.

3. 10 CFR Part 50, Section 50.34a(3) requires that each application to construct a nuclear power reactor provide a description of the provisions for packaging, storage and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

4. 10 CFR Part 50, Section 50.34a(3)(c) requires that each application to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems and (2) a revised estimate of the information required in (b)(2) if the expected releases and exposures differ significantly from the estimate submitted in the application for a construction permit.

5. The Regulatory staff's Safety Evaluation Report and amendments thereto issued prior to the issuance of an operating license contains a description of the radioactive waste systems installed in the nuclear power reactor and a detailed evaluation (including estimated releases of radioactive materials in liquid and gaseous waste and quantities of solid waste produced from normal operation, estimated annual maximum exposures to an individual in the unrestricted area and estimated exposures to the general population) which shows the capability of these systems to meet the appropriate regulations.

6.17 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Senior Manager Nuclear Operations delegates the responsibility for the Radiological Environmental Monitoring Program to the Radiological Control Manager (Figure 6.2-1) or his designated alternate.

The Radiological Control Manager is responsible for administering the offsite Radiological Environmental Monitoring Program. He shall determine that the sampling program is being implemented as described to verify that the environment is adequately protected under existing procedures. He shall also have the responsibility for establishing, implementing, maintaining and approving offsite environmental program sampling, analyses and calibration procedures.

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NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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13 ABSTRACT (200 words or less)

The Beaver Valley Power Station, Unit 2, Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

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